4. Policy Considerations and Environmental Impacts

This Chapter of the Environmental Impact Statement (EIS) describes the policy considerations and potential environmental impacts resulting from each of the management alternatives for implementation of the proposed action and the No Action Alternative. The environmental analysis addresses potential impacts of each alternative on workers, the public, and the environment. The general methodology used throughout this chapter is discussed in Section 4.1.

The policy considerations and environmental impacts of policy alternatives are described in this chapter. One policy alternative is the proposed action, which proposes the adoption of a policy whereby the United States would become involved in the management of the foreign research reactor spent nuclear fuel. The proposed action contains three separate management alternatives for adopting the policy. These management alternatives each contain different implementation alternatives related to that specific management alternative. The second policy alternative is the No Action Alternative which would involve no action by the United States in relation to the foreign research reactor spent nuclear fuel.

Each management alternative would result in very different policy considerations. Much of the foreign research reactor spent nuclear fuel analyzed in this EIS contains highly-enriched uranium (HEU), which can be used to make nuclear weapons. By adopting a policy to manage the foreign research reactor spent nuclear fuel, the proposed action would promote the U.S. goal of nuclear weapons nonproliferation by removing large amounts of HEU from civilian commerce. The No Action Alternative would be in direct conflict with the stated U.S. nuclear weapons nonproliferation goal and would seriously undermine credibility of the United States as a reliable partner in international nuclear weapons nonproliferation activities. Further, foreign research reactor operators may accuse the United States of failing to comply with its obligations under Article IV of the Non-Proliferation Treaty to share the benefits of peaceful nuclear cooperation with other countries.

Each management alternative would also result in very different environmental impacts in the United States which may vary according to the implementation alternatives of each management alternative. The No Action Alternative would have no direct environmental impacts in the United States.

Each of the three management alternatives under the proposed action is briefly summarized here. The three management alternatives were described in greater detail in Chapter 2, Sections 2.2 through 2.4. The policy considerations and environmental impacts of each alternative are described in detail in this chapter.

Management Alternative 1 — Manage Foreign Research Reactor Spent Nuclear Fuel in the United States

Management Alternative 1 of the proposed action entails acceptance and management of the foreign research reactor spent nuclear fuel in the United States. This management alternative would have direct environmental impacts in the United States.

Management Alternative 1 is composed of nine basic implementation components, as well as seven implementation alternatives that alter one of these basic components in some manner. The basic implementation of Management Alternative 1, as well as the seven implementation alternatives, are described in detail in Chapter 2, Section 2.2. The policy considerations and environmental impacts of the basic implementation of Management Alternative 1 are presented in Section 4.2. The policy considerations and environmental impacts of the seven implementation alternatives of Management Alternative 1 are presented in Section 4.3.

Management Alternative 2 — Facilitate the Management of Foreign Research Reactor Spent Nuclear Fuel Overseas

Management Alternative 2 of the proposed action entails U.S. facilitation of overseas management of the foreign research reactor spent nuclear fuel at one or more foreign locations. No foreign research reactor spent nuclear fuel would be accepted into the United States. This would require advance negotiations and agreements with foreign reactor operators, officials in foreign governments, and reprocessing facilities. The outcome of these negotiations is uncertain. This management alternative would have no direct environmental impacts in the United States, unless the Department of Energy (DOE) decides to accept vitrified high-level waste from reprocessing facilities overseas in place of the foreign research reactor spent nuclear fuel. Very few countries have the capability to accept and store high-level wastes (GAO, 1994).

Management Alternative 2 is described in detail in Chapter 2, Section 2.3. Under this management alternative, the United States would negotiate some form of technical assistance and/or financial incentives in return for maintaining some measure of control over the foreign research reactor spent nuclear fuel containing U.S.-origin HEU. The policy considerations and environmental impacts of the two subalternatives of Management Alternative 2 are presented in Section 4.4.

Management Alternative 3 — Combination of Elements from Management Alternatives 1 and 2 (Hybrid Alternative)

Management Alternative 3 entails some combination of the elements from Management Alternatives 1 and 2, and is referred to as the Hybrid Alternative. Management Alternative 3 would likely have more direct environmental impacts in the United States than Management Alternative 2, but less than Management Alternative 1.

Management Alternative 3 is described in detail in Chapter 2, Section 2.4. For purposes of analysis, a sample Hybrid Alternative has been included to demonstrate one possible combination of elements within Management Alternatives 1 and 2, and to allow an analysis of its impacts. It is important to note that the Hybrid Alternative described is merely an example for analysis purposes, and is only one of numerous possible combinations of elements from Management Alternatives 1 and 2.

Under the Hybrid Alternative described, DOE and the Department of State would facilitate the reprocessing of the foreign research reactor spent nuclear fuel at western European reprocessing facilities (i.e., Dounreay, United Kingdom or Marcoule, France) for foreign research reactor operators in countries that can accept the reprocessing waste, as in Management Alternative 2. DOE would accept and manage the remaining foreign research reactor spent nuclear fuel in the United States, as in Management Alternative 1. The policy considerations and environmental impacts of the sample Hybrid Alternative (Management Alternative 3) are described in Section 4.5.

Other Alternatives and Comparisons

The No Action and Preferred Alternatives are discussed in Sections 4.6 and 4.7, respectively. Comparisons across all the alternatives of the potential impacts and costs are presented in Section 4.8 and 4.9, respectively. Finally, this chapter concludes by comparing the risks due to the alternatives to the risks due to other common activities in Section 4.10.

4.1 Overview of Environmental Impacts

4.1.1 Presentation of the Environmental Impacts

Potential environmental impacts associated with each segment of the affected environment of the proposed action are addressed in this chapter. These segments are presented in this section in the following order:

- Marine transport impacts,
- Port of entry impacts,
- Ground transport impacts, and
- Management Site impacts.

The impact analyses of these four segments are described in more detail in Appendices C, D, E, and F, respectively. Effects of each implementation alternative of Management Alternative 1 of the proposed action on U.S. nuclear weapons nonproliferation goals and objectives are also discussed. In addition, this chapter summarizes the potential costs associated with the alternatives. Details on costs are presented in Appendix F.

Spent nuclear fuel is transported in strong, heavy casks (NRC, 1993). After the spent nuclear fuel is delivered, the empty casks must be transported back on a return trip. Under most of the alternatives, empty casks would be transported overland, through U.S. ports, and on ships. There would be minor nonradiological impacts (vehicle emissions and potential traffic accidents) during ground transport of empty casks. These nonradiological ground transport impacts are included as part of the assessment in this EIS.

4.1.2 Key Assessment Factors

A key assessment factor is one that may differentiate among alternatives, has a measurable impact, or be of public interest. The detailed analysis of potential environmental impacts presented in the appendices of this EIS did not reveal any factor likely to cause a large impact. Because radiation exposure and its consequences is a topic of great public interest, emphasis is placed upon exposure to radiation, although DOE considers the evaluated effects of radiation to be small.

During handling operations, the principal hazard would come from radiation being emitted by the foreign research reactor spent nuclear fuel. Without adequate shielding, the radiation levels at the surface of some of the spent nuclear fuel itself would often be high enough to induce a prompt fatality. This radiation can and would be attenuated (i.e., reduced) by the shielding materials of the transportation cask, such as lead, steel, and polyethylene. Further, since radiation intensity decreases with distance, maintaining a distance from the cask would also provide radiation protection. At 100 m (330 ft) from the cask, the radiation levels would not be detectable above background radiation. All foreign research reactor spent nuclear fuel handling at the proposed foreign research reactor spent nuclear fuel management sites would take place at considerable distances from the public (greater than 100 m or 330 ft). Recently, actual radiation measurements were taken by the State of North Carolina, Department of Environment, Health, and Natural Resources, of the casks used in the first shipment of the 153 spent fuel elements covered by the Urgent Relief Environmental Assessment (DOE, 1994m). In every case, the State of North Carolina reported detecting no radiation above background levels (radiation exposure from natural sources) at a distance of 1 meter (3.3 ft) from the package surface (State of North Carolina, 1994).

Accidents involving foreign research reactor spent nuclear fuel could potentially also result in releases of radioactive material which could cause radiation exposures. For most accidents, essentially none of the radioactive material would be released because it is an integral part of the solid fuel. Larger quantities of radioactive elements could be released only when the accident generates enough energy to release particles of foreign research reactor spent nuclear fuel to the atmosphere, such as with a fire. However, the probability of such accidents is very small. For most accidents, the energy would not be high enough to damage the foreign research reactor spent nuclear fuel, so that none of the radioactive material would be released.

4.1.3 General Radiological Health Effects

The effect of radiation on people depends upon the kind of radiation exposure (alpha and beta particles, and gamma and x-rays) and the total amount of tissue exposed to radiation. The amount of radiant energy imparted to tissue from exposure to ionizing radiation is referred to as absorbed dose. The sum of the absorbed dose to each tissue, when multiplied by certain quality and weighting factors that take into account radiation quality and different sensitivities of these various tissues, is referred to as effective dose equivalent (EDE).

An individual may be exposed to radiation from outside the body, or from inside the body because radioactive materials may enter the body by ingestion or inhalation. External dose is different from internal dose in that it is delivered only during the actual time of exposure. An internal dose, however, continues to be delivered as long as the radioactive source is in the body (although both radioactive decay and elimination of the radionuclide by ordinary metabolic processes decrease the dose rate with the passage of time). The dose from internal exposure is calculated over 50 years following the initial exposure.

The annual radiation dose limit to the public from nuclear facilities operated by DOE is 100 mrem per year (NRC, 1991). The potential foreign research reactor spent nuclear fuel management sites covered by DOE operations normally operate such that the public's dose is undetectable. For comparison, it is estimated that the average individual in the United States receives a dose of about 350 mrem per year from all sources combined, including natural and medical sources of radiation and radon. A modern chest x-ray, for example, results in an approximate dose of 8 mrem, while a diagnostic hip x-ray results in an approximate dose of 83 mrem (DOE, 1995c).

Radiation can also cause a variety of adverse health effects in people. A large dose of radiation can cause prompt death. At low doses of radiation, the most important adverse health effect for depicting the consequences of environmental and occupational radiation exposures (which are typically low doses) is the potential inducement of cancers that may lead to death in later years. This effect is referred to as latent cancer fatalities (LCF) because the cancer may take years to develop and for death to occur, and may never actually be the cause of death.

In addition to LCF, other health effects could result from environmental and occupational exposures to radiation. These effects include nonfatal cancers among the exposed population and genetic effects in subsequent generations. Table 4-1 shows the dose-to-effect factors for these potential effects as well as for LCF. For simplicity, this EIS presents estimated effects of radiation only in terms of LCF. The nonfatal cancers and genetic effects are less probable consequences of radiation exposure, and are less serious.

Table 4-1 Risk of LCF and Other Health Effects from Exposure to Radiation

Population[a]	*LCF*[b]	*Nonfatal Cancers*	*Genetic Effects*	*Total Detriment*
Workers	0.0004	0.00008	0.00008	0.00056
Public	0.0005	0.0001	0.00013	0.00073

[a] *The difference between the worker risk and the general public risk is attributable to the fact that the general*

population includes more individuals in sensitive age groups (that is, less than 18 years of age and more than 65 years of age).

[b] *When applied to an individual, units are lifetime probability of LCF per rem of radiation dose. When applied to a population of individuals, units are excess number of cancers per person-rem of radiation dose. Genetic effects as used here apply to populations, not individuals.*

The collective or "population" dose to an exposed population is calculated by summing the estimated doses received by each member of the exposed population. This is referred to as a "population dose." The total population dose received by the exposed population is measured in person-rem. For example, if 1,000 people each received a dose of 0.001 rem, the population dose would be 1.0 person-rem (1,000 persons x 0.001 rem = 1.0 person-rem). The same population dose (1.0 person-rem) would result if 500 people each received a dose of 0.002 rem (500 persons x 0.002 rem = 1 person-rem).

The factor used in this EIS to relate a dose to its effect is 0.0004 LCF per person-rem for workers and 0.0005 LCF per person-rem for individuals among the general population (DOE, 1995c). The latter factor is slightly higher because of some individuals in the public, such as infants, who may be more sensitive to radiation than workers. These factors are based on the *1990 Recommendations of the International Commission on Radiological Protection* (ICRP, 1991), and are consistent with those used by the U.S. Nuclear Regulatory Commission (NRC) in its rulemaking *Standards for Protection Against Radiation* (NRC, 1991). The factors apply where the dose to an individual is less than 20 rem and the dose rate is less than 10 rem per hour. At doses greater than 20 rem, the factors used to relate radiation doses to LCF are doubled. At much higher doses, prompt effects, rather than LCF, may be the primary concern. Unusual accident situations that may result in high radiation doses to individuals are considered special cases. No such cases are expected with either incident-free handling or accidents with foreign research reactor spent nuclear fuel.

These concepts may be applied to estimate the effects of exposing a population to radiation. For example, if 100,000 people were each exposed only to background radiation (0.3 rem per year), 15 LCF per year would be expected (100,000 persons x 0.3 rem per year x 0.0005 LCF per person-rem = 15 LCF per year).

Sometimes, calculations of the number of LCF associated with radiation exposure do not yield whole numbers and, especially in environmental applications, may yield numbers less than 1.0. For example, if 100,000 people were each exposed to a total dose of only 1 mrem (0.001 rem), the population dose would

These same concepts apply to estimating the effects of radiation exposure on a single individual. Consider the effects, for example, of exposure to background radiation over a lifetime. The "number of LCF" corresponding to a single individual's exposure to 0.3 rem per year over a (presumed) 72-year lifetime is:

> 1 person x 0.3 rem per year x 72 years x 0.0005 LCF per person-rem = 0.011 LCF or one chance in 91 of an LCF.

Again, this should be interpreted in a statistical sense; that is, the estimated effect of background radiation exposure on the exposed individual would produce a 1.1 percent chance that the individual would incur a latent fatal cancer. Alternatively, this method estimates that about 1 person in 91 would die of cancers induced by background radiation.

4.1.4 Risks

Another concept important to the presentation of results in this EIS is the concept of risk. Risks are most important when presenting accident analysis results. The chance that an accident might occur during the conduct of an operation is called the probability of occurrence. An event that is certain to occur has a probability of 1.0 (as in 100 percent certainty). If an accident is expected to happen once every 50 years, the frequency of occurrence is 0.02 per year (1 occurrence every 50 years = 0.02 occurrences per year). A frequency estimate can be converted to a probability statement. If the frequency of an accident is 0.02 per year, the probability of the accident occurring in a 10-year program is 0.2 (10 years x 0.02 occurrences per year).

Once the frequency (occurrences per year) and the consequences (for radiation effects, measured in terms of the number of LCF caused by the radiation exposure) of an accident are known, the risk can be determined. The risk per year is the product of the annual frequency of occurrence times the number of LCF. This annual risk expresses the expected number of LCF per year, taking account of both the annual chance that an accident might occur and the estimated consequences if it does occur.

For example, if the frequency of an accident were 0.2 occurrences per year and the number of LCF resulting from the accident were 0.05, the risk would be 0.01 LCF per year (0.2 occurrences per year x

Assessment (DOE, 1994m), are presented in Appendix F, Section F.5. The average of these measurements is 2.3 mrem per hour at 1 m (3.3 ft) from the surface of the cask. Recent measurements taken by the State of North Carolina on foreign research reactor spent nuclear fuel shipment packages, covered by the Urgent Relief Environmental Assessment, showed that the external dose rate at 1 m (3.3 ft) was undetectable above background radiation levels (State of North Carolina, 1994).

To be conservative, the analyses in this chapter use the regulatory limit of 10 mrem per hour at 2 m (6.6 ft) from the side of the transport vehicle for the radiation dose rate near the foreign research reactor spent nuclear fuel casks. This conservative value was used in the calculations of incident-free doses to members of the public, marine transport workers, port workers, and ground transport workers. For radiation workers at the spent nuclear fuel management sites, the dose rate in the vicinity of the casks was estimated by the conservative methodology presented in Appendix F, Section F.5.

4.1.6 The Effects of Radiation on Plants and Animals

There is no convincing evidence from the scientific literature that chronic radiation doses below 1 rad per day will harm animal or plant populations. It is highly probable that limitation of the exposure of the most exposed humans (the critical human group, living on and receiving full sustenance from the local area) to 100 mrem per year will lead to dose rates to plants and animals in the same area of less than 1 rad per day. DOE and NRC regulations limit annual human exposures to values far lower than those that have caused observable damage in plant and animal populations. Therefore, specific radiation protection standards for nonhuman biota are not needed (IAEA, 1992).

4.2 Management Alternative 1 – Manage Foreign Research Reactor Spent Nuclear Fuel in the United States – Basic Implementation

This section presents the policy considerations and potential environmental impacts of the basic implementation of Management Alternative 1. Under the basic implementation of Management Alternative 1, all the foreign research reactor spent nuclear fuel could be accepted into the United States. DOE and the Department of State believe this would promote the nuclear weapons nonproliferation objective of reducing, and ultimately eliminating, civil commerce in HEU. The spent nuclear fuel could be managed safely and securely at any of five DOE sites.

Policy Considerations

A critical result of this basic implementation of Management Alternative 1 would be the continued viability and vitality of the Reduced Enrichment for Research and Test Reactors (RERTR) Program, which has the goal of minimizing and eventually eliminating the use of HEU in civil nuclear programs. The successful development of alternative fuels for research reactors and the expansion of the program to Russia, the other Newly Independent States, China, South Africa, and other countries, and the establishment of a world-wide norm discouraging the use of HEU, is dependent on a United States commitment to action. Finally, this basic implementation of Management Alternative 1 would support the Administration's nuclear weapons nonproliferation objective of not encouraging reprocessing for either nuclear power or nuclear explosive purposes.

Another crucial consideration associated with Management Alternative 1 is the *Treaty on the Non-Proliferation of Nuclear Weapons*. The parties to the Non-Proliferation Treaty met in May of 1995 and agreed to extend the treaty indefinitely and without conditions. One key to the success of the 1995

Non-Proliferation Treaty parties that the nuclear weapons states had complied with their obligations under Article IV of the Non-Proliferation Treaty to assist the non-nuclear weapons states with peaceful applications of nuclear energy.

Although the Non-Proliferation Treaty was extended indefinitely, the parties also agreed to review the treaty every five years to ensure that all parties are in compliance. Any country which has been compelled to shut down its research reactors could accuse the United States of not having complied with its treaty obligations. This accusation, however ill-founded, could be made not only by the affected countries, but by any country opposed to the interests of the United States.

The amount of foreign research reactor spent nuclear fuel that would be accepted under the basic implementation of Management Alternative 1 is up to approximately 19.2 metric tons of heavy metal (MTHM) representing approximately 22,700 elements. This amount is an upper limit because if some nations were to reprocess their research reactor spent nuclear fuel, for example, the amount of foreign research reactor spent nuclear fuel accepted into the United States would be reduced. Under the basic implementation of Management Alternative 1, approximately 4.6 metric tons (5.1 tons) of HEU would be removed from international commerce.

4.2.1 Marine Transport Impacts

Because the basic implementation of Management Alternative 1 involves ocean transport, DOE and the Department of State considered the environmental impacts on the global commons (i.e., portions of the ocean not within the territorial boundary of any nation) in accordance with Executive Order 12114 (U.S. Federal Register, 1979).

4.2.1.1 General Assumptions and Analytic Approach

The basic implementation of Management Alternative 1 includes the shipment of approximately 837 transportation casks containing foreign research reactor spent nuclear fuel over a 13-year period. Of these, approximately 721 transportation casks would be transported by sea to the United States, with the remainder (116) coming overland from Canada. DOE would prefer to consolidate the approximately 721 casks on board ships to minimize the number of voyages, but it is also possible that approximately 721 voyages could be required. This section evaluates the impacts of the marine transportation, including shipment in international waters from the port of origin to the United States and coastal shipping in United States territorial waters.

Four types of commercial cargo ships are considered to be candidates to carry foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1: containerized, breakbulk (general cargo), roll-on/roll-off, and purpose-built vessels (see Appendix C for a more complete description of these vessels). DOE and the Department of State assumed that all casks would be transported in standard International Standards Organization 20-ft shipping containers, because this is current shipping practice.

Nonradiological impacts associated with the marine shipment of 721 containerized transportation casks would be minimal. The United States receives more than 56,000 ships engaged in foreign trade at its ports each year (DOC, 1994). Shipping an additional 56 containers per year on average over the 13-year receipt period is not likely to cause any additional ships to sail beyond the number already scheduled. In the event that chartered vessels are used for this program, up to 10 voyages per year could be required, which is only 0.02 percent of the number engaged in regular commerce. Additional nonradiological impacts would be very small whether chartered or regularly scheduled commercial vessels are used. The number of containers handled on a regular basis is so large that the addition of the foreign research reactor spent nuclear fuel containers would add essentially no impacts (cargo vessels typically carry 800 to 1,000 containers per voyage). While nonradiological marine events such as unloading or cargo shifting accidents would be possible, the nonradiological impacts would be miniscule.

The radiological impacts of transporting the foreign research reactor spent nuclear fuel by sea were considered in two ways, incident-free impacts and accident impacts. The incident-free impacts would be those that occur simply due to the marine shipping of foreign research reactor spent nuclear fuel, assuming there are no accidents. The ship's crew would be the affected individuals in this case. The accident impacts would be the consequences of reasonably foreseeable accidents that might occur. These two evaluations are discussed in the following two sections, with additional details in Appendix C.

4.2.1.2 Conservative Assumptions and Maximum Estimated Impacts of Incident-Free Marine Transport

The primary impact of incident-free marine shipping of foreign research reactor spent nuclear fuel would be upon the crews of the ships used to carry the spent nuclear fuel casks. Since the crew of a ship is normally separated from the cargo and shielded by both the cargo and the ship's structure, the risk to the crew from spent nuclear fuel transport during most crew activities would be extremely low (DOE, 1994m). The exceptions would include the exposure to the crew during loading and off-loading of the spent nuclear fuel ISO containers and during daily inspection of the ship's cargo, including the containers housing the spent nuclear fuel transportation casks. Therefore, the crew exposure during loading, daily inspection, and unloading of the transportation casks has been incorporated into the incident-free marine transport analysis. The exposure to dock workers at the foreign research reactor spent nuclear fuel port of entry is assessed in Section 4.2.2.

Daily inspections of the casks is the activity that would result in the largest doses to the ship's crew, with the inspectors considered the maximally exposed workers during incident-free marine transport. For any given voyage, DOE and the Department of State conservatively assumed that the same three inspectors would conduct all of the inspections. The impact on the inspectors would be a function of the number of inspections performed, which would depend upon the amount of time the cask is onboard. Therefore, the incident-free radiological impact on the inspectors would depend upon the total duration of the voyage, including days at sea, in intermediate ports, and days in coastal sailing between intermediate ports. The duration of the voyage was selected as the weighted average of the duration of all the shipments necessary for 721 transportation casks. (See Appendix C for further details regarding this assumption.)

To maximize the estimated impact from incident-free transport, DOE and the Department of State made conservative assumptions regarding crew exposure. Specifically, DOE and the Department of State conservatively assumed that eight and two casks (loaded two casks per hold) would be shipped per voyage of chartered and regularly scheduled commercial ships, respectively. This assumption would result in additional exposure of the ship's crew due to the effect of loading casks into holds where a loaded cask would have already been stowed, and would also increase the exposure to the crew members performing daily inspections. The additional exposure would be a result of the combination of the radiation fields surrounding each of the transportation casks.

Assuming 56 casks per year, the number of annual voyages required would range from 7 to 28, depending upon the number of casks per ship. Although the foreign research reactor spent nuclear fuel would be shipped from 40 countries worldwide and to both U.S. coasts over a 13-year receipt period, DOE and the Department of State conservatively assumed that a single crew could be involved in up to 9 voyages per back. Additionally, to determine the dose to the maximally exposed worker in the ship's crew, DOE and the Department of State conservatively assumed that the same individuals would conduct all the daily onboard inspections.

The dose received during daily cargo inspection would be a major contributor to the crew dose, so the duration of the voyage is an important consideration. Chartered vessels would sail directly to the port(s) of entry, yielding an average voyage duration of 18 days. DOE and the Department of State conservatively assumed that all shipments aboard regularly scheduled commercial breakbulk vessels would include two intermediate port stops in the United States, which would add 3 days to the voyage.

Table 4-2 presents the maximum estimated incident-free marine transport doses and risks. Values are provided for a chartered ship (which would not make intermediate port calls) and for a regularly scheduled commercial vessel. The values are based on the estimated time the cask would be onboard multiplied by the dose per day received as a result of inspections, plus the crew dose due to the foreign research reactor spent nuclear fuel container loading and off-loading activities. While the use of a chartered ship would result in higher per-shipment impacts (eight casks per shipment versus two for regularly scheduled commercial ships), the reduced number of voyages would offset this increase in per-shipment impacts exposed worker risk is 0.00052 LCF, which is based on the annual regulatory limit every year for 13 years. This means that the chance of this individual incurring a cancer due to the basic implementation of Management Alternative 1 would be less than one in a thousand.

[illegible]

4.2.1.3 Conservative Assumptions and Maximum Estimated Impacts of Accidents During Marine Transport

The basic implementation of Management Alternative 1 could potentially impact the marine environment in the event of an accident involving the release of radioactive material from the spent nuclear fuel. This section discusses possible accidents and their consequences.

The range of accidents that could occur during marine transport is quite broad. The ship could collide with another vessel or an object such as a shoal, rock, or wreck. Foul weather could damage or sink the ship, or the ship could experience a fire, explosion, or other problem. To reduce the risk due to potential accidents, the casks that would carry the foreign research reactor spent nuclear fuel have been designed to prevent damage to the cask contents in all but the most severe, and least likely, cases. See Appendix B of this EIS for a description of the foreign research reactor spent nuclear fuel transportation casks.

Two scenarios emerge that could potentially threaten the marine environment and possibly humans: the cask could be damaged and then involved in a fire, or the cask could sink. These cases are discussed in more detail below.

Cask Damaged Followed by a Fire

A ship carrying foreign research reactor spent nuclear fuel could be involved in a severe collision with another ship. It is possible that a transportation cask, carried on a ship involved in such a collision, could be exposed to impact forces resulting from the collision. In that event, the cask could be damaged. However, only a small fraction, at most, of the force generated in a collision of one ship with another

The limiting accident is a ship collision severe enough to breach a cask carrying foreign research reactor spent nuclear fuel and also cause a large fire. Some of the radioactive contents of the cask could be released and carried into the air by the heated gases of the fire as a plume of radioactive particles. For an airborne release of this type to occur, the cask-carrying vessel must stay afloat during and immediately after the accident. In practice, this would mean that the ship must stay afloat for a period of some hours following an accident of the requisite severity. This latter condition must be satisfied for atmospheric dispersal to occur, even though marine casualty files indicate that a common outcome of severe ship collisions is rapid sinking, often within a matter of minutes. Assuming the cask was damaged by a severe collision; and the ship remained afloat despite the severe collision; and the cask was engulfed in flames for a time sufficient to release a radioactive plume, there would likely be no human population on the ocean (excluding the crew) who could be affected.

It is possible that the ship could be in coastal waters (i.e., beyond the port's sea buoy) at the time of this severe collision. Except in port, a ship is seldom within 16 km (10 mi) of a population center, so the port accident public risk analysis in the next section covers public risk in this scenario. The ship's crew and people onboard other vessels that may come to provide assistance could be exposed to any released radioactive material. The number of people potentially exposed would be less than that used in the port accident analysis for populations near a port [less than 1.6 km (1 mi) from the port]. Additionally, accident frequencies at sea tend to be lower than in-port accident frequencies. Therefore, both the consequences and risks for an accident at sea are covered by the results of the port accident analysis.

Risks associated with this type of accident at sea are covered by the risks of the same type of accident in ports because humans in the vicinity of the accident at sea are much fewer in number than even the least populated port.

Sunken Cask

The second scenario of concern is that a foreign research reactor spent nuclear fuel cask or casks would be sunk. This could be the result of the ship sinking, of the casks being somehow swept overboard, or of a ground transport accident on a causeway. Submersion of an intact cask would not necessarily result in a release of its contents, as spent nuclear fuel casks are designed to withstand at least a 15 m (50 ft) immersion. It has been demonstrated that cask seals will remain intact at much greater depths (DOE, 1994m). Should a loaded foreign research reactor spent nuclear fuel cask (damaged or undamaged) sink anywhere in the U.S. coastal waters, it will be recovered regardless of depth. U.S. Coastal waters in this case refers to waters within the 12 mile territorial limit. Recovery would be accomplished, even in the deepest parts of U.S. coastal waters, such as in Puget Sound, which reaches 305 meters or 1,000 feet (Encyclopedia Americana, 1991). Elsewhere in the world, spent nuclear fuel casks can, and likely would, be recovered from water up to 200 m (660 ft) deep, which is beyond the range typical of coastal and port depths. Typically 200 m (660 ft) is considered the limit of the continental shelf. Recovery at depths greater than 200 m (660 ft) is possible but is more difficult.

If a sunken cask containing foreign research reactor spent nuclear fuel were recovered, the effect on the marine environment would be minimal, even if the recovery effort required up to 1 year to complete. The release to the ocean water of radioactive particles from the spent nuclear fuel requires that first the metallic spent fuel corrode, then the radioactive particles escape from the cask. Even if the cask were damaged, the most likely damage to a spent nuclear fuel cask, either from mechanical trauma or excessive depth, would be failure of the seal. Seal failure would allow seawater to enter the cask to begin the corrosion of the metallic spent nuclear fuel, but the flow of water through the cask to carry out the radioactive material would be minimal due to the small cross sectional area of the failed seal. The decay heat from the spent nuclear fuel is low, thereby providing no driving force to expel water out of the cask through the failed seal.

If a cask was not recovered, the radioactive constituents of spent nuclear fuel would be released slowly over time into the surrounding waters. Some of the radioactive material would be removed from the water by adhesion to suspended sediments. Assuming a cask were submerged on the deep ocean bottom and not recovered, the peak human dose to an individual ingesting seafood harvested from the area in which the breached submerged spent nuclear fuel cask would be located would be 114 mrem per year. If a sunken cask in coastal waters was not recovered, the peak human dose is conservatively estimated to be 14,000 mrem per year. Consequences to humans and to marine biota are presented in Table 4-3. Other studies of similar circumstances indicate that the individual dose would be even lower (DOE, 1980). Uranium (the major constituent of the spent nuclear fuel) has been found not to bioaccumulate in fish, and bioaccumulates only slightly in crustaceans and mollusks (IAEA, 1976). The peak doses for humans, fish, crustaceans, and mollusks are presented in Table 4-3 in the situation where a chartered ship carrying eight casks might sink in deep ocean. Doses for humans and other animals are expressed in units of rem and rad, respectively. Rem is discussed in some detail in Section 4.1.3. While rem is only used for measuring human exposure to radiation, rad is used to measure exposure of nonhumans to radiation. Rad is a unit of absorbed dose from ionizing radiation.

The probability provided in Table 4-3 is the probability of one ship accident and loss of a cask during the entire program. The consequences are from one unrecovered cask. The program risk is the product of the probability and the consequences. Humans would not be the principally exposed species in a marine accident involving foreign research reactor spent nuclear fuel. Estimates were made of the dose to the biota received from a damaged cask containing foreign research reactor spent nuclear fuel. This analysis assumes that the cask would lay on the deep ocean floor where it would slowly release its radioactive inventory whether it was damaged in the collision or not.

Table 4-3 Impacts of Unrecovered Casks in Deep Ocean

	Probability	*Consequences*	*Program Risk*
MEI (human)	1.7×10^{-6}	114 mrem/yr	0.00019 mrem/yr
Fish	1.7×10^{-6}	640 rad/yr	1.1 mrad/yr
Crustaceans	1.7×10^{-6}	880 rad/yr	1.4 mrad/yr
Mollusks	1.7×10^{-6}	30,000 rad/yr	49 mrad/yr

Risks associated with the release of the contents of the spent nuclear fuel elements into the deep ocean are expected to be very small due to the low probabilities and limited consequences. The highest estimated risk to the MEI is 0.00019 mrem per year for every year that the cask leaks and this hypothetical individual ingests seafood harvested from near the cask. DOE and the Department of State assume that these conditions could apply for about 5 years, so the total MEI dose would be 0.00095 mrem. This translates into a maximum estimated MEI risk of 5×10^{-10} LCF. This means that this hypothetical individual's additional chance of incurring an LCF would be less than one in a billion. The risks to fish, crustaceans, and mollusks are low enough that no adverse impacts would be expected.

Probabilities, consequences, and risks were also calculated for the cases of unrecovered casks in coastal waters, both undamaged and damaged. The results are presented in Table 4-4, again in terms of rem for humans and rad for other animals. In coastal waters, cask recovery is considered likely (NEA, 1988), which makes the probabilities in Table 4-4 low. Comparing Tables 4-3 and 4-4 shows that the consequences of a sunken cask in coastal waters would be greater than in the deep ocean, but when multiplied by the probabilities, the risks are actually lower.

Table 4-4 Impacts of Unrecovered Casks in Coastal Waters

	Probability One Undamaged Cask	*Consequences One Undamaged Cask*	*Program Risk*
MEI (human)	2.3 x 10^{-8}	190 mrem/yr	4.3 x 10^{-6} mrem/yr
Fish	2.3 x 10^{-8}	77 mrad/yr	1.8 x 10^{-6} mrad/yr
Crustaceans	2.3 x 10^{-8}	81 mrad/yr	1.9 x 10^{-6} mrad/yr
Mollusks	2.3 x 10^{-8}	210 mrad/yr	4.8 x 10^{-6} mrad/yr
	Probability One Damaged Cask	*Consequences One Damaged Cask*	*Program Risk*
MEI (human)	4.6 x 10^{-11}	14,000 mrem/yr	6.4 x 10^{-7} mrem/yr
Fish	4.6 x 10^{-11}	620 mrad/yr	2.9 x 10^{-8} mrad/yr
Crustaceans	4.6 x 10^{-11}	660 mrad/yr	3.0 x 10^{-8} mrad/yr
Mollusks	4.6 x 10^{-11}	14,000 mrad/yr	6.4 x 10^{-7} mrad/yr

These risk estimates were derived assuming that the foreign research reactor spent nuclear fuel is shipped at a rate of one cask per voyage. Assuming a different shipping schedule, such as eight casks per voyage, would not result in a different estimate of the risks. The potentially higher consequences of an accident involving more than one shipping cask would be balanced by the reduced probability of an accident due to the reduced number of shipments. For example, the risk associated with one shipment of eight casks is equivalent to the risks associated with eight single cask shipments.

4.2.1.4 Marine Transport Cumulative Impacts

The cumulative impact of radioactive material shipments on ships' crews beyond that discussed in Section 4.2.1.2 was not estimated. In estimating the cumulative impact on port workers (see the following section) it was possible to estimate the total number of shipments of radioactive material through a port. However, it is not as simple to estimate the total number of shipments of radioactive material that involve the same ship and crew. It is expected that each ship's crew would be exposed to fewer of the shipments of radioactive material than that assumed for the port worker in the cumulative impact analysis for the port. For port workers, the impacts of the shipments other than the foreign research reactor spent nuclear fuel were of the same order of magnitude, but lower than the foreign research reactor spent nuclear fuel shipments. Therefore, the individual crew member's exposure from shipments other than the foreign research reactor spent nuclear fuel shipments would be a small fraction of the dose received due to the foreign research reactor spent nuclear fuel shipments.

4.2.1.5 Marine Transport Mitigation Measures

The principal environmental impact that would occur during marine transport would be radiation dose to the ships' crews. Most of this dose occurs because crew members must visually inspect the cargo every day for safety reasons, and the inspections cannot be curtailed.

The magnitude of the estimated impacts from this portion of the basic implementation of Management Alternative 1 is primarily due to two items: the conservative assumption that the radiation field emanating from all of the casks would be at the regulatory limit (as opposed to the levels of one-tenth of the regulatory limit that have been observed in past foreign research reactor spent nuclear fuel shipments), and the conservative assumption that the same crew member is involved in inspections for all of the casks on nine shipments during any given year. In reality, neither of these conservative assumptions would be likely to occur. Nevertheless, to ensure that no member of a ship's crew could receive a dose above what is allowed for a member of the general public, DOE would mitigate this effect by implementing a system through its shipping contractor to track each ship and crew involved in the shipment of foreign research reactor spent nuclear fuel. DOE would also include a clause in the contract for shipment of the foreign research reactor spent nuclear fuel requiring that other crew members be used if any crew member approaches a 100 mrem dose in any year.

If a cask or casks were sunk in deep ocean or coastal waters, DOE and the Department of State would employ modern underwater search techniques to locate and recover the cask(s), thus minimizing the potential impacts to marine life.

4.2.2 Port Activities Impacts

4.2.2.1 General Assumptions and Analytic Approach

To assess the range of potential impacts on ports at which a ship carrying f[illegible] nuclear fuel might call, 13 ports of entry representing a wide range of port city population densities were selected for detailed evaluation. Eight of the ports—Charleston, SC; Elizabeth, NJ (for the New York City area); Philadelphia, PA; Norfolk, VA (representing Hampton Roads); Jacksonville, FL; Savannah, GA; Wilmington, NC; and Military Ocean Terminal at Sunny Point (MOTSU), NC—are East Coast ports that represent high, medium, and low population density ports. The Norfolk Terminal was selected to

- Partial unloading of cargo,
- Partial reloading of cargo, and
- Port exit to the sea buoy.

As with the marine transport, the port impacts were evaluated for two conditions: incident-free and accident conditions. Summary results are presented in the following sections. Details of the analysis are presented in Appendix D.

4.2.2.2 Conservative Assumptions and Maximum Estimated Impacts of Incident-Free Port Activities

As stated in Section 2.6, no spent nuclear fuel transportation cask has ever released its contents (radioactive material), even as a result of an accident. For this reason, release of radioactive material is not considered as part of the incident-free analysis. The only impact considered is that caused by radiation exposure due to radiation emitted by foreign research reactor spent nuclear fuel contained within the transportation casks. Since no radioactive material would be released, there would be no impacts on land, water, or air quality in any of the ports or any of the waterways used by ships in the transport of foreign research reactor spent nuclear fuel.

Risks associated with the foreign research reactor spent nuclear fuel in incident-free conditions in port are predominantly those to inspectors and port workers. Port workers and inspectors are not radiation workers as defined by NRC regulations. Thus, the maximum allowable annual exposure for these personnel would be 100 mrem, the same radiation dose limit established by the NRC to protect individual members of the public (DOE, 1990c). When a ship arrives in its first port, the spent nuclear fuel package would be inspected by customs officials, U.S. Coast Guard personnel, and others. Up to six inspections, estimated at up to 15 minutes per person per spent nuclear fuel cask, were conservatively assumed. Once inspections are complete, the ship would partially unload and reload cargo. After that, DOE and the Department of State conservatively assumed that the ship would proceed to another intermediate port and then to the port of entry for the foreign research reactor spent nuclear fuel.

To determine the incident-free risks associated with port operations, two types of ships were considered for the shipment of the foreign research reactor spent nuclear fuel. In the first case, DOE and the Department of State conservatively assumed that all shipments were made on regularly scheduled commercial breakbulk ships. This type of vessel was selected because it maximized the time required for port activities, such as off-loading and inspections. In addition, during the operations at the intermediate port stops, DOE and the Department of State conservatively assumed that other unloading and loading operations would occur in the vicinity of the container with the loaded foreign research reactor spent nuclear fuel cask in one of the intermediate ports. Risks associated with these activities, which are comparable to the risks associated with the off-loading of the foreign research reactor spent nuclear fuel, have been included in the assessment. Transport of the material on this type of vessel would therefore result in the highest worker radiation doses in the incident-free analysis. All worker exposures were calculated by estimating the times required for activities and the distances from the transportation cask to where the worker would most likely be located.

To provide a measure of the difference in the worker exposures resulting from the use of cargo vessels other than the regularly scheduled commercial breakbulk vessels, the analysis was also performed for port operations associated with the use of a chartered container vessel. This type of vessel requires the least amount of time to unload. DOE and the Department of State also assumed that a chartered vessel would not make any intermediate port stops, so that the ship's port of entry into the United States would also be the port of entry for the foreign research reactor spent nuclear fuel. Use of these two types of vessels in the analysis provides an estimate of the range of the maximum incident-free risk associated with port operations.

At the port of entry, the casks would be off-loaded by port workers, and arrangements would be made for the immediate departure of the foreign research reactor spent nuclear fuel from the port. In recognition of instances where some delay may occur, DOE and the Department of State conservatively assumed a delay of up to 24 hours in a secure staging area. The 24-hour period for the staging of spent nuclear fuel casks was selected because it is possible that, on occasion, the spent nuclear fuel casks would not leave the secure staging area the same day that they arrived, depending on variables such as the time of day the casks clear customs and the weather. Nonetheless, DOE and the Department of State consider it unlikely that the casks would remain in the staging area for longer than 24 hours.

To estimate the maximum individual exposure, the shipments were divided into East Coast and West Coast shipments, depending on the country of origin. Spent nuclear fuel shipments from Europe, Africa, the Middle East and parts of South America were designated as East Coast shipments, all others were designated as West Coast shipments. Under these assumptions, the East Coast port(s) would receive approximately 535 casks and the West Coast port(s) approximately 186 casks. DOE and the Department of State also conservatively assumed for this analysis that all the shipments would pass through the same intermediate ports as the shipments on regularly scheduled commercial vessels and have the same port of entry.

Further, DOE and the Department of State made the very conservative assumption that the same inspectors and workers would handle every cask shipment. The per-shipment doses were then multiplied by the number of shipments for the East Coast to determine the maximally exposed worker dose for the basic implementation of Management Alternative 1.

In determining the worker population exposure, all shipments (East Coast and West Coast) were considered. This results in the integrated dose for the entire basic implementation of Management Alternative 1 which would span 13 years. The maximum estimated incident-free risks to port personnel due to the basic implementation of Management Alternative 1 are presented in Table 4-5. The incident-free risk to the general public would be zero because only workers would be near the casks in port.

This table shows the maximally exposed worker dose, worker population dose, and associated risks for the shipment of foreign research reactor spent nuclear fuel as containerized cargo on a regularly scheduled commercial breakbulk vessel and as cargo on a chartered container vessel. These figures represent the range of maximum estimated impacts for the various shipping modes available for the ocean transport of foreign research reactor spent nuclear fuel.

As the table shows, the highest estimated maximally exposed worker risk is 0.00052 LCF, which is based on the annual regulatory limit every year for 13 years. This means that the chance of this individual incurring a cancer due to the basic implementation of Management Alternative 1 would be less than one in a thousand.

The highest total population risk for port workers is 0.012 LCF, which is much less than one LCF.

Table 4-5 Incident-Free Port Activity Impacts[a,b]

Impacts per Cask Transfer

Risk Group	Regularly Scheduled Commercial Breakbulk Ship: Maximally Exposed Worker Dose (mrem)	Maximally Exposed Worker Risk (LCF)	Population Dose to Workers per Cask (person-rem)	Population Risk (LCF)	Chartered Container Ship: Maximally Exposed Worker Dose (mrem)	Maximally Exposed Worker Risk (LCF)	Population Dose to Workers per Cask (person-rem)	Population Risk (LCF)
Inspectors	3.8	0.0000015	0.013	0.0000052	1.3	5×10^{-7}	0.0053	0.0000021
Port Handlers, Intermediate Ports	2.2	9×10^{-7}	0.018	0.0000071	----	----	----	----
Port Handlers, Port of Destination	2.0	8×10^{-7}	0.0066	0.0000026	0.46	1.8×10^{-7}	0.0015	6×10^{-7}
Port Staging Personnel	0.36	1.4×10^{-7}	0.0045	0.0000018	0.4	2×10^{-7}	0.0046	0.0000018
Maximum	3.8	0.0000015	----	----	1.3	5×10^{-7}	----	----
Total	----	----	0.042	0.000017	----	----	0.011	0.0000045

Impacts for the Entire Basic Implementation

Risk Group	Regularly Scheduled Commercial Breakbulk Ship: Maximally Exposed Worker Dose (mrem)	Maximally Exposed Worker Risk (LCF)	Population Dose to Workers (person-rem)	Population Risk (LCF)	Chartered Container Ship: Maximally Exposed Worker Dose (mrem)	Maximally Exposed Worker Risk (LCF)	Population Dose to Workers (person-rem)	Population Risk (LCF)
Inspectors	1,300[c]	0.00052[c]	9.4	0.0038	670	0.00027	3.8	0.0015
Port Handlers, Intermediate Ports	1,186	0.00047	13	0.0052	----	----	----	----
Port Handlers, Port of Destination	1,072	0.00043	4.8	0.0019	250	0.0001	1.1	0.00044
Port Staging Personnel	190	0.000076	3.2	0.0013	210	0.000084	3.3	0.0013
Maximum	1,300[c]	0.00052	----	----	670	0.00027	----	----
Total	----	----	30	0.012	----	----	8.2	0.0032

[a] *These results are based on the assumption that the dose rates associated with the casks are all based on the regulatory limit. Historically, the average of these dose rates has been equal to about one-tenth of the regulatory limit, so this assumption is conservative.*

[b] *These results are all based on the assumption that each voyage carries two casks. This assumption is conservative because chartered ships may carry up to eight casks.*

[c] *With all the conservative assumptions in this analysis, the maximally exposed worker dose could theoretically exceed the annual regulatory limit. Therefore, DOE would require mitigation measures to keep the maximally exposed worker dose down to 100 mrem per year or lower. These results are based on the conservative assumption that one individual receives the maximum annual dose (100 mrem) every year for 13 years. See Appendix D for maximally exposed worker doses without mitigation measures.*

4.2.2.3 Conservative Assumptions and Maximum Estimated Impacts of Accidents During Port Activities

Section 4.2.1.3 discussed the impacts of marine accidents that could occur either in the open ocean or during coastal passages. This section discusses the impacts of accidents that could occur anywhere from the sea buoy into the port and at the pier.

Methodology

An analysis of reasonably foreseeable accidents must evaluate the consequences of possible accidents and the probability of an accident occurring. In incident-free marine transport, some exposure would be expected from radiation emitted from the casks. In the case of accidents, the probability of exposure is only an estimate of a hypothetical event. Accident probabilities are derived from published maritime accident rates. The analysis of ship collisions concludes that only one hold of the ship carrying the foreign research reactor spent nuclear fuel transportation casks would be subject to sufficient forces to potentially result in cask damage. There is no difference between the risks associated with a single shipment with two casks in a hold, and two shipments of a single cask each. The consequences of the accident with two casks in the hold may be as large as twice the consequences of an accident involving one cask. But the probability of an accident involving the ship carrying the two casks is half the probability of one of the two ships carrying a single cask being involved in an accident. Therefore, the potential risk from accidents, marine transportation of spent nuclear fuel was modeled in the port accident analysis as occurring in one cask per shipment.

Because accidents can be of any degree of severity, from a "fender bender" to one involving severe impact and prolonged fire, the severity spectrum is divided into a number of accident severity categories. Each category is assigned a conditional probability of occurrence [i.e., a probability (given that an accident occurs) that it will be of that particular severity]. In general, the more severe the accident, the more remote the chance of such an accident. In this analysis, the accident severity spectrum is divided into six categories (Wilmot et al., 1981), which are discussed further in Appendix D. The accident scenarios considered in this analysis fall into the three most severe of the six severity categories.

Accidents in the first three, least severe, categories result in no release of material from the spent nuclear fuel transportation cask. These categories include all the accident scenarios associated with handling the spent nuclear fuel cask, including dropping the cask during transfer from the vessel to the truck or train. The transportation casks are certified to maintain their integrity when dropped from 15 m (50 ft) onto a perfectly unyielding surface. During the cask transfer, however, the crane may lift the cask higher than 15 m (50 ft). As the dock surface is softer than "perfectly unyielding," the soft surface of the dock would compensate for the greater drop height. Studies (DOE, 1994m) have shown that a cask can be dropped from much higher than the certification test height onto a yielding surface, without breaching.

The accidents analyzed in the three highest severity categories include collision of vessels, either in the approach to the harbor or when the vessel transporting the foreign research reactor spent nuclear fuel would be docked. The category 4 accident severity category models a collision of two vessels resulting in the breach of the transportation cask. Severity categories 5 and 6 model collisions that would breach the cask and subsequent fires that would cause the release of additional material, with category 6 fires being more intense than those for category 5.

As mentioned above, the spectrum of accidents, including a container breach and fire, were evaluated at two locations in each of the 13 ports of entry selected to envelop the port impacts. The approach to each port, from the sea buoy to the selected dock, was examined to determine the location where the accident would be most likely as well as have the largest consequence. This point is typically near the highest population center along the approach to the pier, and DOE and the Department of State conservatively selected this point for accident analysis. The second location where the spectrum of accidents was assumed to occur is at pier-side.

At these two locations, the probability of an accident was assigned, based on historical ship accident data (see Appendix D for details). These data were used to develop accident frequencies for collisions between vessels large enough to generate the forces sufficient to damage the cask (additional details on the development of the model used are provided in Appendix D), and to develop the frequency of collisions concurrent with fires (Lloyds, 1991). These data include information on a large number of ship voyages

Ports Selected for Accident Analyses

Analyses of the impacts of possible accidents at representative ports were conducted. Thirteen ports were selected as being representative of the full range of ports in the United States, based on population and geography. Three of the ports are high-population density ports, two on the East Coast (Elizabeth, NJ and Philadelphia, PA) and one on the West Coast (Long Beach, CA). Five of the ports (Portland, OR; Tacoma, WA; Concord NWS, CA; Jacksonville, FL; and Norfolk, VA) are medium-population density ports, three on the West Coast and two on the East Coast. The remaining ports (MOTSU, NC; Galveston, TX; Savannah, GA; Wilmington, NC; and Charleston, SC) are low-population density ports. The 13 potential ports of entry for which accidents were analyzed collectively have a range of populations and geography that ensure that the results of these analyses are representative of the results that would have been reached if the analyses had been conducted for all ports. Additionally, these 13 ports include all 10 of the ports that meet all of the port selection criteria.

To demonstrate the representative nature of the analyses performed, a plot was made of the analyzed accident consequences for mean meteorological conditions at each port versus the port's population in a 16-km (10-mi) radius (Figure 4-1). The analyzed data points are shown as dots. The straight line represents the linear least squares fit of the data. Since the straight line represents an average of the data, some deviation from the line for individual data points is expected. The data fit well, with a correlation factor of 0.994455 (a correlation factor of 1.0 implies a perfect fit). This plot demonstrates the expected increase in the total population dose with an increase in port population. Deviations from the line by the calculated data are typically due to the distribution of population in relation to the local meteorology. Where most of the population is downwind of the port in normal weather, the corresponding population dose would likely be above the average line. For comparison, the total population dose due to background radiation is shown in the upper right corner. This comparison shows that population dose resulting from a severe accident would be approximately 0.2 percent of the annual background population dose.

As a check that the data from the 16-km (10-mi) radius population is valid, a similar analysis was performed correlating the 80-km (50-mi) radius population and accident consequences for seven ports. This analysis confirmed that the population dose as a function of population is linear, and therefore confirms that the range of ports selected for accident analysis fully covers the entire range of U.S. ports. More specific discussion of the results of the analyses is provided in Appendix D. This linearity of consequences and population show that any port selected for use as an intermediate port or port of entry for the foreign research reactor spent nuclear fuel, ranging from the least populous port (MOTSU) to the most populous port (Elizabeth) and including major ports of intermediate population, has had representative accident analyses performed.

Probabilities of Port Accidents

The probability of an accident occurring can be determined from historical statistics on ship collisions and mishaps. Maritime accident rate data from a Lloyd's of London database covering approximately 900,000 port calls by commercial vessels over a 15-year period (1978 to 1993) were examined to develop accident probabilities. The data indicate that the basic accident rate in and near ports is slightly less than 0.0001 accidents per port transit, or approximately 1 accident per 10,000 port visits.

Only the most severe accidents, however, would cause any damage to the cask. Thus, the conditional probabilities of occurrence of each accident severity were also developed from this database. As discussed in Appendix D, a conditional probability is defined as the probability, given that an accident has occurred, that it will be of a certain severity. To calculate overall probability of an accident of a particular severity, the base accident probability (accident rate) must be multiplied by the conditional probability.

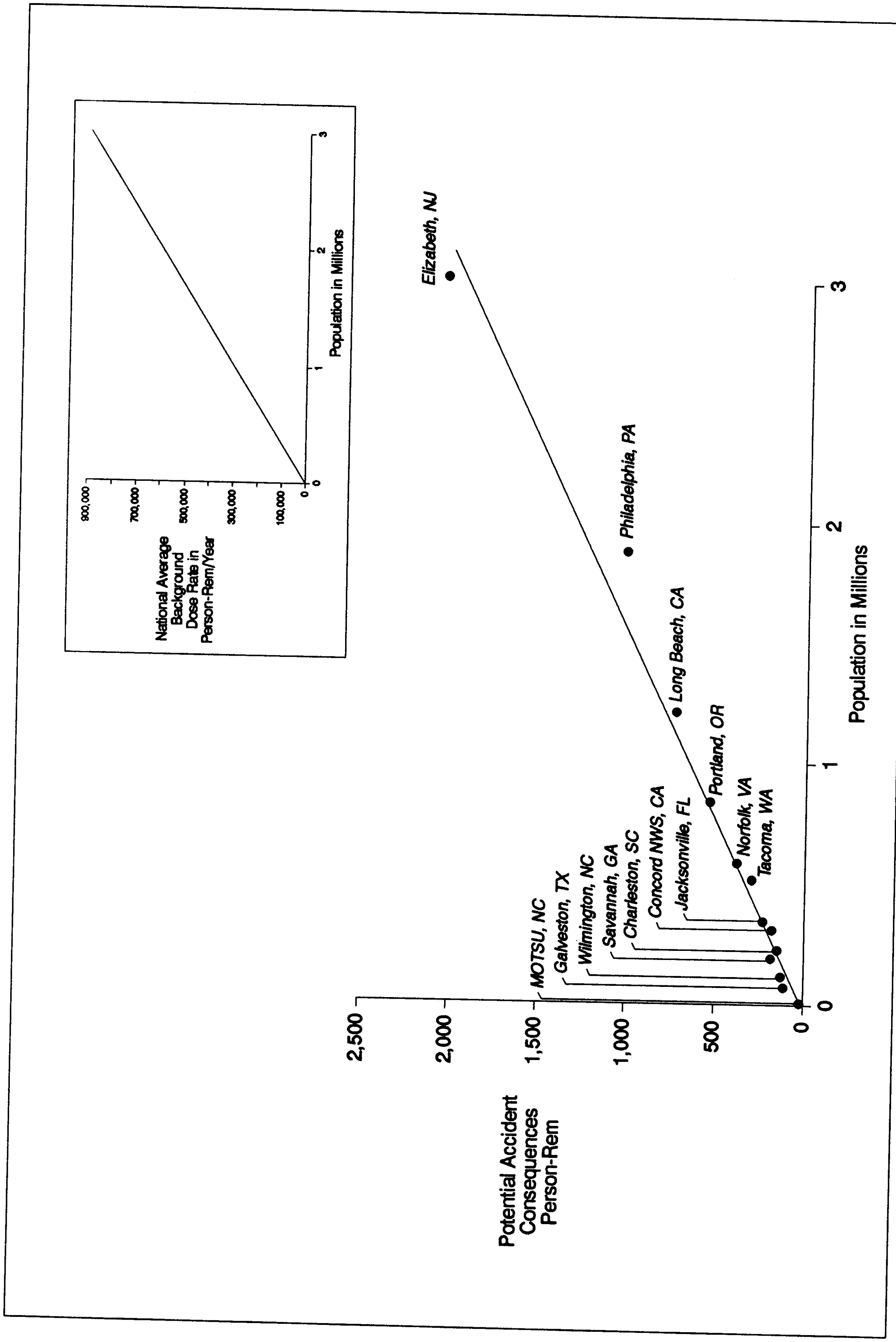


Figure 4-1 Consequences Versus Population [for a 16-km (10-mi) Radius]

Accidents are ranked according to their release categories. Release category 4 would result from the cask being damaged and compromised. Release category 5 would result from a damaged and compromised cask being enveloped in a fire. Release category 6 would result from a damaged and compromised cask being enveloped in a longer fire than a category 5 fire. The probabilities for the category 4, 5, and 6 accidents are 0.000006, 5×10^{-9}, and 6×10^{-10}, respectively. DOE and the Department of State assumed that it was equally likely that the accident occurs at the dock or in the channel, during the approach to the dock.

Consequences of Port Accidents

The consequence of an accident indicates the result, given that the accident were to occur, without any consideration for the likelihood of the accident occurring. The analysis conducted to determine the impacts of an accident involving foreign research reactor spent nuclear fuel in ports yields two different measures of the consequences. One measure is a calculation of the number of LCF that might result if the accident were to occur. These results are presented in Table 4-6 for the three most severe types of accidents under mean meteorological conditions.

The results presented in Table 4-6 are based on the mean consequences, so they are equivalent to results expected for the accidents in the respective release categories. These results are also based on the conservative assumption that accidents involve a cask carrying the highest inventory of nuclear material expected. Appendix D provides information on the consequences associated with the range of spent nuclear fuel types considered.

Examination of Table 4-6 shows that the most adverse consequence (2.9 LCF) arises from a Release category 5 accident in the channel approaching the Port of Elizabeth. This places the vessel just west (and generally upwind) of New York City. Although some of the release fractions change between categories 5 and 6, most of them do not. Therefore, the total population dose and the related number of LCF are about the same for Release categories 5 and 6. Release category 4 would be a release with no fire. In the absence of a fire, the release would remain at ground or water level without wide dispersion, hence the greatly reduced number of affected individuals and reduced consequences.

In addition to calculating the health effects of an accident on man, the MACCS code also calculates the impact of the accident on the land and structures around the accident site. These effects are characterized by the costs of activities required to bring the land and structures back into a usable condition. These activities are characterized as (1) no remedial action required; (2) decontamination – the resources can be returned to use immediately after clean-up; (3) interdiction – the resources must be temporarily abandoned, for several years, prior to their return to use; and 4) condemnation – the resources are considered unusable for an extended period. In all of the consequence analyses performed for each of the accident sites, there are no costs calculated that are associated with decontamination, interdiction, or condemnation. This means that all of the land and structures would be immediately available for use. (The consequences calculated by MACCS for the immediate vicinity of the accident are based on average value for the area within 1.6 km (1 mi) of the accident. Even though the average consequences calculated by MACCS show no costs associated with accident clean-up, the area immediately around the ship carrying the foreign research reactor spent nuclear fuel (i.e., the dock area) may require some remedial activity).

A sensitivity analysis was performed to address the potential impact of shipboard fires with extremely high temperatures that could result in the foreign research reactor spent nuclear fuel attaining temperatures above the melting point of the aluminum-based fuel or the combustion temperature of the TRIGA fuel. This analysis shows that the maximum consequences of such a fire are a factor of 100 larger than those

Table 4-6 Port Accident Consequences (LCF)

Locations	Accident Severity Category[a]		
	4	5	6
Elizabeth at the Dock	0.00010	2.8	2.7
Elizabeth in the Channel	0.00016	2.9	2.8
Long Beach at the Dock	0.000093	2.0	2.0
Long Beach in the Channel	0.000035	1.8	1.9
Philadelphia at the Dock	0.000078	1.2	1.2
Philadelphia in the Channel	0.000037	1.2	1.2
Portland at the Dock	0.000034	0.52	0.53
Portland in the Channel	0.000023	0.50	0.51
Norfolk at the Dock	0.000024	0.38	0.37
Norfolk in the Channel	0.000013	0.30	0.30
Charleston at the Dock (Wando Terminal)	0.000011	0.19	0.19
Charleston at the Dock (NWS Charleston)	0.0000068	0.22	0.22
Charleston in the Channel	0.000017	0.19	0.19
Tacoma at the Dock	0.000024	0.75	0.80
Tacoma in the Channel	0.000017	0.63	0.66
Concord NWS at the Dock	0.000019	0.90	0.96
Concord NWS in the Channel	0.000041	1.40	1.50
Jacksonville at the Dock	0.000012	0.31	0.31
Jacksonville in the Channel	0.000011	0.24	0.25
Savannah at the Dock	0.000025	0.23	0.23
Savannah in the Channel	0.0000059	0.18	0.19
Wilmington at the Dock	0.000017	0.22	0.23
Wilmington in the Channel	0.0000042	0.098	0.10
Galveston at the Dock	0.000032	0.64	0.70
Galveston in the Channel	0.000014	0.63	0.69
MOTSU at the Dock	0.0000032	0.099	0.11
MOTSU in the Channel	0.0000042	0.098	0.10

[a] *These accident release categories are the three highest in severity.*

calculated for the base case (Appendix D, Section D.5.4.2.2, Table D-31). An extremely high temperature ship fire is highly unlikely (one in ten billion per shipment) and the risks are comparable to those calculated in the base case. This analysis also addressed the impact of an accident on the land around the port. Using the same characterizations as described in the preceding paragraph, the largest mean impact distance is a decontamination and interdiction distance which is limited to approximately 300 m (1,000 ft). The analysis also shows that this distance is representative of the impact of this highly improbable type of accident at any of the ports included in the proposed action.

Risks

The calculated risk (probability times consequence) to the nearby population on a per-shipment basis assuming one cask per shipment and for the entire basic implementation of Management Alternative 1 is presented in Table 4-7. Each risk value is the sum of the risks from accident severity categories 4, 5, and 6. (A sensitivity study was performed to assess the risks associated with accidents that result in extremely high temperature fires. This sensitivity study was limited to an analysis of the per-shipment risks associated with shipment of spent nuclear fuel through the highest population density port, Elizabeth, NJ. Even though the consequences of this type of an accident are orders of magnitude larger than those calculated for the base case analysis, this type of event is highly improbable and the risks are comparable to those calculated in the base case. A more detailed comparison of the base case and sensitivity analyses is presented in Appendix D, Section 5.4.3.2.)

Table 4-7 Port Accident Risks

Port	*Per Shipment of One Cask*		*Total All Shipments*	
	Population Dose (person-rem)	*Population Risk (LCF)*	*Population Dose (person-rem)*	*Population Risk (LCF)*
Elizabeth via:				
• Two High Population Ports	0.00013	5.6×10^{-8}	0.070	0.000029
• One High and One Intermediate Population Port	0.00011	4.8×10^{-8}	0.060	0.000025
• One High and One Low Population Port	0.00011	4.5×10^{-8}	0.057	0.000024
• Two Intermediate Population Ports	0.000056	2.4×10^{-8}	0.030	0.000013
• One Intermediate and One Low Population Port	0.000051	2.2×10^{-8}	0.027	0.000011
• Two Low Population Ports	0.000046	2.0×10^{-8}	0.024	0.000010
• Direct	0.000042	1.8×10^{-8}	0.022	0.0000094
Long Beach via:				
• Two High Population Ports	0.00011	4.7×10^{-8}	0.058	0.000025
• One High and One Intermediate Population Port	0.000080	3.4×10^{-8}	0.042	0.000018
• One High and One Low Population Port	0.000071	3.0×10^{-8}	0.038	0.000016
• Two Intermediate Population Ports	0.000050	2.1×10^{-8}	0.026	0.000011
• One Intermediate and One Low Population Port	0.000041	1.8×10^{-8}	0.022	0.0000092
• Two Low Population Ports	0.000032	1.4×10^{-8}	0.017	0.0000072
• Direct	0.000028	1.2×10^{-8}	0.015	0.0000062
Philadelphia via:				
• Two High Population Ports	0.00011	4.5×10^{-8}	0.057	0.000024
• One High and One Intermediate Population Port	0.000088	3.7×10^{-8}	0.047	0.000020
• One High and One Low Population Port	0.000083	3.5×10^{-8}	0.044	0.000019
• Two Intermediate Population Ports	0.000031	1.4×10^{-8}	0.016	0.0000072
• One Intermediate and One Low Population Port	0.000026	1.1×10^{-8}	0.014	0.0000061
• Two Low Population Ports	0.000021	9.3×10^{-9}	0.011	0.0000049
• Direct	0.000017	7.5×10^{-9}	0.0092	0.0000040
Portland via:				
• Two High Population Ports	0.000090	3.8×10^{-8}	0.047	0.000020
• One High and One Intermediate Population Port	0.000059	2.5×10^{-8}	0.031	0.000013
• One High and One Low Population Port	0.000050	2.2×10^{-8}	0.027	0.000011
• Two Intermediate Population Ports	0.000029	1.3×10^{-8}	0.015	0.0000066
• One Intermediate and One Low Population Port	0.000020	9.0×10^{-9}	0.011	0.0000047
• Two Low Population Ports	0.000011	5.1×10^{-9}	0.0059	0.0000026
• Direct	0.0000073	3.2×10^{-9}	0.0039	0.0000017
Norfolk via:				
• Two High Population Ports	0.000095	4.0×10^{-8}	0.050	0.000021
• One High and One Intermediate Population Port	0.000076	3.2×10^{-8}	0.040	0.000017
• One High and One Low Population Port	0.000071	3.0×10^{-8}	0.037	0.000016
• Two Intermediate Population Ports	0.000019	8.3×10^{-9}	0.0098	0.0000044
• One Intermediate and One Low Population Port	0.000014	6.1×10^{-9}	0.0072	0.0000032
• Two Low Population Ports	0.0000088	4.0×10^{-9}	0.0046	0.0000021
• Direct	0.0000048	2.1×10^{-9}	0.0025	0.0000011
Charleston (Wando Terminal) via:				
• Two High Population Ports	0.000092	3.9×10^{-8}	0.049	0.000021
• One High and One Intermediate Population Port	0.000074	3.1×10^{-8}	0.039	0.000016
• One High and One Low Population Port	0.000069	2.9×10^{-8}	0.036	0.000015
• Two Intermediate Population Ports	0.000016	7.4×10^{-9}	0.0087	0.0000039

The manner of evaluating the per-shipment risk of LCF in Table 4-7 is the same as for the per-shipment population exposure risk. Once again, shipping the foreign research reactor spent nuclear fuel through or into high-population density ports would increase the risk, as would using ships that pass through intermediate ports on their way to the port of entry.

The range of total population risks would be from 0.070 to 0.00069 person-rem for the population dose and from 0.000029 to 3.2×10^{-7} LCF for the risk, comparing shipping to Elizabeth via two high-population density ports and shipping to MOTSU without intermediate ports. The highest estimated population risk due to port accidents that might occur due to the basic implementation of Management Alternative 1 is 0.000029 LCF. This means that there would be less than a one in ten thousand chance of some member of the public incurring an LCF due to the basic implementation of Management Alternative 1 port transits.

The highest estimated MEI accident risk is conservatively determined by multiplying the accident probability by the consequences, in terms of dose to the MEI, of that accident. The MEI in this case is assumed to be an individual at the center of the plume less than 1.6 km (1 mi) from the accident. The highest average MEI doses calculated for the accident severity categories are: 0.11 mrem for category 4, 117 mrem for category 5, and 95 mrem for category 6. See Appendix D, Section D.5.4.2.2 for details. The reason MEI dose for category 6 is relatively lower than that for category 5 is because the larger category 6 associated fire would disperse the radioactive material faster and farther than the category 5 fire. For the 721 shipments in the basic implementation of Management Alternative 1, and using the per port transit accident probabilities in Appendix D, the highest MEI accident risk is estimated to be 0.00042 mrem. This corresponds to about 2×10^{-10} LCF. This means that the chance of the MEI incurring an LCF due to a port accident under the basic implementation of Management Alternative 1 would be less than one in a billion.

Emergency Management and Response

Emergency response capabilities for a foreign research reactor spent nuclear fuel mishap would be available through the U.S. Coast Guard and the local jurisdictions surrounding each candidate port of entry, with specialized support available from DOE. The specialized analysis and identification of potential hazards, use of the robust "Type B" packaging, specific emergency plan and procedure development, training, response rehearsal, and interagency coordination for efficient and effective response would minimize the potential consequences should a foreign research reactor spent nuclear fuel mishap occur. The specific emergency management and response capabilities and responsibilities are described in Chapter 2, Section 2.7.

At military ports, the U.S. Coast Guard routinely provides safety/security screen escorts. The addition of foreign research reactor spent nuclear fuel shipments would have almost no effect on their ongoing operations.

Consequences of Port Accidents

A sensitivity analysis was performed to address the potential impact of extremely high temperature fires, fires that could result in the foreign research reactor spent nuclear fuel attaining temperatures above the melting point of the aluminum based fuel or the combustion temperature of the TRIGA fuel, on the consequences of an accident in port. This analysis, which uses the Port of Elizabeth, NJ as the site of the accident, is presented in Appendix D, Section D.5.4.3.2, and shows that even though the consequences of this type of an accident are two orders of magnitude larger than those calculated for the base case analysis, this type of event is highly improbable and the risks are comparable to those calculated in the base case.

This analysis also addressed the impact of an accident on the land around the port. Using the same characterizations as described in the preceding paragraph, the largest mean impact distance is a decontamination and interdiction distance which is limited to approximately 300 meters (1000 feet). The analysis also shows that this distance is representative of the impact of this highly improbable type of accident at any of the ports included in the proposed action.

4.2.2.4 Cumulative Impacts of Port Activities

Port workers are expected to be exposed to other shipments of radioactive materials in addition to those associated with the basic implementation of Management Alternative 1. These shipments include DOE and commercially initiated programs. An assessment has been made of the cumulative impact of the incident-free dose to the maximally exposed worker from all of these activities. The cumulative analysis is based on data collected at several ports for 2.5 years (January 1992 through June 1994). The maximally exposed port worker is estimated to receive less than 10 mrem per year from commercial shipments. Details of this analysis are presented in Appendix D, Section D.4.6. As previously stated, based on cask dose rates equal to the regulatory limit, the maximally exposed port worker could receive an annual dose greater than the NRC and DOE regulatory limit of 100 mrem per year (NRC, 1991). Therefore, DOE would implement mitigation measures.

4.2.2.5 Port Activities Mitigation Measures

As with marine transport, the principal environmental impact that would occur during port activities is radiation dose to workers. No members of the general public would be close enough to the transportation cask to receive any radiation dose. The workers would receive this dose during safety inspections and handling activities which cannot be curtailed.

Two conservative assumptions in this analysis drive the maximally exposed worker dose higher than would actually be expected. The radiation dose rate near every foreign research reactor spent nuclear fuel shipping container is assumed to be equal to the regulatory limit and the same individual is assumed to conduct all the inspections. Neither of these is actually likely to occur.

Nevertheless, DOE and the Department of State would require, through a clause in the shipping contracts, some administrative controls on the port workers to mitigate the radiation doses to the workers during inspection and handling activities. DOE and the Department of State would implement a system to track the inspectors and other port workers actually involved in the shipment of foreign research reactor spent nuclear fuel. If any inspector's or worker's dose approaches 100 mrem in any year, then DOE and the Department of State would require other inspectors or workers to be used. In this way, the maximally exposed worker dose would be constrained to the regulatory limit.

If a cask or casks were sunk in coastal waters, DOE and the Department of State would employ modern underwater search techniques to locate and recover the cask(s), thus minimizing the potential impacts to marine life.

4.2.2.6 Environmental Justice at the Port(s)

Executive Order 12898 deals with the issue of environmental justice and directs Federal agencies to identify and address, as appropriate, disproportionately high and adverse human health or environmental effects of their programs, policies, and activities on minority and low-income populations.

The concept of environmental justice is discussed in more detail in Appendix A. During normal port activities associated with receipt of the foreign research reactor spent nuclear fuel shipments—including harbor activities, unloading the ship, transfer of the spent nuclear fuel containers to truck or train, and movement out of the port city—the dominant radiological impacts have been shown to be the exposures received by the workers in the immediate vicinity of the shipping container. These individuals include the inspectors, shipping container handlers, truck drivers, etc. Since the intensity of the gamma radiation falls off rapidly with distance, the doses that might be received by other workers and members of the general population can in theory be calculated, but would not generally be measurable or distinguishable from natural background radiation.

Potential radiological impacts to people residing near the port are associated with low probability (less than one in a million) accidents that are so severe that the spent nuclear fuel casks would be ruptured and a fire would burn long enough around the cask that some of the radioactive material would be released. In this case, some of the radioactive spent nuclear fuel might be vaporized and lifted by the heat of the fire and carried downwind of the accident location. Where and how far this radioactive material would go before being deposited on the ground would depend on how high the heat from the fire lofts it and the particular weather conditions at the time. Most of this vaporized spent nuclear fuel would be expected to be deposited in the first few kilometers downwind of the fire but small amounts could be carried out for several tens of kilometers.

Because the particular details of both the accident conditions (such as the severity of a fire) and the weather conditions at the time of an accident could vary so much, a range of accident conditions and wind directions, wind speeds, and other weather conditions were examined during the evaluation of accidents (see Section 4.2.2.3). Population impact evaluations were performed for distances out to 80 km (50 mi). The risk of LCF was found to be so small that zero LCF would be expected due to accidents at ports.

Appendix A describes minority populations and low-income households residing near the ports. Calculations for incident-free and accident conditions clearly demonstrate that for the general population, including minority and low-income groups, the radiological impacts would be very low. Minority or low-income populations living near the potential ports of entry would not be subjected to any greater impacts. Therefore, these populations would not receive disproportionately high and adverse impacts. They would be subject to the same very low impacts as would the general population.

Implementation of the proposed action would have extremely low nonradiological effects on the environment at candidate ports, including the social and economic status of the general population, minority populations, and the low-income population surrounding candidate ports. Economic benefits that would result from increased cargo handling and transportation in the port area would be extremely small for the general population or any particular segment of the population residing near candidate ports.

4.2.3 Ground Transport Impacts

Foreign research reactor spent nuclear fuel is transported in large, heavy containers called transportation casks. Transportation casks are designed and constructed to contain the radioactivity in spent nuclear fuel during severe transportation accidents. NRC has estimated that transportation casks will withstand 99.4 percent of truck and rail accidents without sustaining damage sufficient to breach the transportation cask (NRC, 1987). Only in the worst conceivable conditions, which are of low probability, could a transportation cask of the type used to transport spent nuclear fuel be so damaged that there is a reasonable possibility of release of radioactivity to the environment.

Spent nuclear fuel has been transported along highways, railways, and waterways since 1949. Federal standards describe the routing requirements for spent nuclear fuel shipments. Spent nuclear fuel transported includes foreign research reactor, commercial, naval, and DOE spent fuel. Since 1949, there have been 21 incidents involving vehicles carrying irradiated fuel elements. None of these incidents resulted in damage to the structural integrity of a cask or the release of the cask's contents.

4.2.3.1 Conservative Assumptions and Analytic Approach

Transportation impacts may be divided into two parts: the impacts due to incident-free transportation and the impacts due to transportation accidents. For incident-free transportation and transportation accidents, impacts may be further divided into two parts: nonradiological impacts and radiological impacts. The nonradiological impacts consist of the vehicular impacts of transportation, such as vehicular emissions and traffic accidents.

For incident-free transportation, the radiological impacts would result from the radiation field that surrounds the cask. For transportation accidents, the radiological impacts would be based on the radioactivity released from the spent nuclear fuel transportation cask during the accident. Impacts are estimated for workers and the population along the transportation route.

For both incident-free transportation and transportation accidents, methodology developed by NRC and used by DOE in the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (SNF&INEL Final EIS) (DOE, 1995c) was used to estimate the impacts for foreign research reactor spent nuclear fuel in this EIS. These impacts were quantified as the estimated traffic accidents. The probabilities and the magnitudes of exposure are discussed in Appendix E. Incident-free risk factors were calculated for crew and public exposure to radiation emanating from the cask, and public exposure to the chemical toxicity of the transportation vehicle exhaust. The probability of incident-free exposure and the magnitudes of exposure are discussed in Appendix E.

Calculation of risk factors was accomplished by first using the HIGHWAY (Johnson, et al., 1993a) and INTERLINE (Johnson, et al., 1993b) computer codes to choose representative routes in accordance with the U.S. Department of Transportation regulations. These codes provide population estimates along the routes so that the RADTRAN (Neuhauser, 1993) and RISKIND (Yuan, et al., 1993) codes could be used to determine the risk factors associated with ground transportation activities. These computer codes are described in more detail in the Programmatic SNF&INEL Final EIS (DOE, 1995c) and Appendix E of this EIS.

The single largest contributor to the ground transport population doses (about 80 percent) calculated with RADTRAN was found to be the dose to members of the public at truck stops. The parameters used to calculate doses during truck stops are quite conservative. The parameters are based on the assumption that stops occur as a function of distance, with a truck stop rate of 0.011 hr per km (0.018 hr per mi). This stop rate results in over an hour of stop time per 100 km (62 mi) of travel. It was further assumed that at each stop, an average of 50 people are exposed at a distance of 20 m (66 ft). These parameters were used because they are the default parameters in the RADTRAN code and they were used in the Programmatic transportation activity. The risk factors are generally a function of distance and total population along the port to management site route, so the port chosen often shifted between Phase 1 and Phase 2. Conversely, the lower-bound case assumes ports with the lowest risk factors.

The average case is designed to provide a realistic estimate of the ground transport risk of transporting the foreign research reactor spent nuclear fuel. The risk factors are an arithmetic average of the risk factors for all acceptable ports. This represents the risk associated with the basic implementation of Management Alternative 1 and receiving foreign research reactor spent nuclear fuel at a variety of commercial ports.

Since each potential port of entry and each management site is capable of receiving spent nuclear fuel via rail or highway, the program was analyzed using each mode of transportation. The exception to this is the Nevada Test Site which has no existing rail capability, so that link was approximated by a hypothetical rail line to the Yucca Mountain Site. Additionally, the potential to use trucks to carry the relatively small casks from ports to potential foreign research reactor spent nuclear fuel management sites and rail to carry larger casks between potential foreign research reactor spent nuclear fuel management sites was analyzed. Site to site shipment would not occur until approximately 2006, so it is difficult to precisely predict which cask would be used. The analysis is based on a truck cask that carries 4 times as much spent nuclear fuel as a foreign cask, and a rail cask that carries 10 times as much spent nuclear fuel.

4.2.3.2 Impacts of Incident-Free Ground Transport

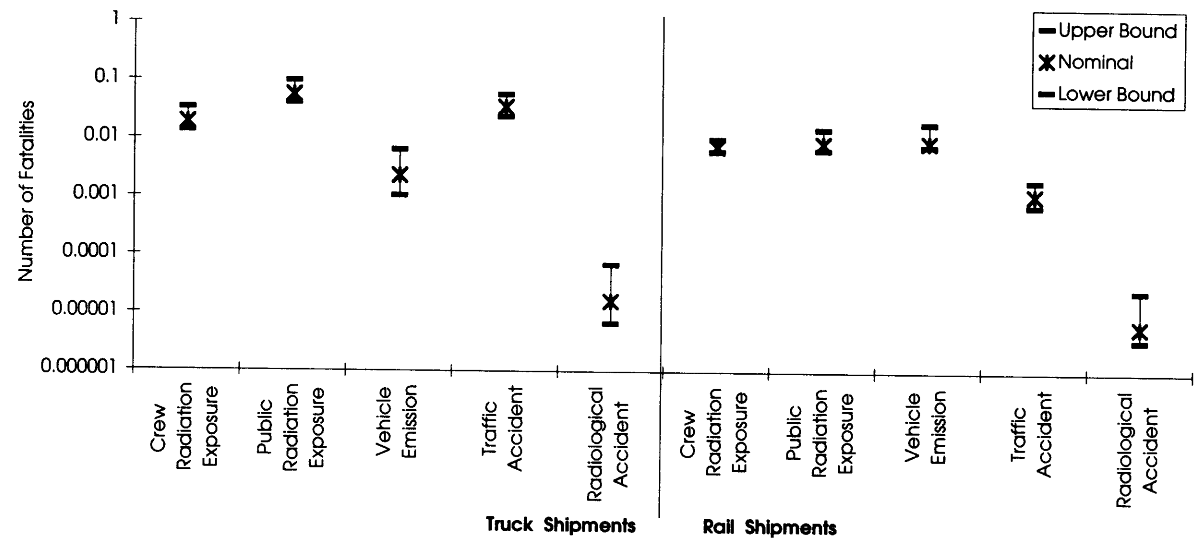
The incident-free transportation of spent nuclear fuel was estimated to result in total population risk that ranged from 0.013 to 0.30 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the transportation workers. Thus, the calculated maximum risk value for overland transportation is less than one fatality from cancer due to the basic implementation of Management Alternative 1. The range of fatality estimates is caused by two factors: (1) the option of using truck or rail to transport spent nuclear fuel; and (2) combinations of Phase 1 and Phase 2 sites that create varying cask shipment numbers and distances.

The estimated number of LCF due to radiation exposure for transportation workers ranged from 0.006 to 0.071. The estimated number of radiation-related LCF for the general population ranged from 0.007 to 0.22, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.001 to 0.052. These incident-free results apply to the workers and the public because both would be close enough to the cask to receive some radiation dose.

The impacts of transportation which are based on four Programmatic SNF&INEL Final EIS (DOE, 1995c) programmatic alternatives are summarized in Figures 4-2 through 4-5. The impacts of these additional programmatic alternatives are described in more detail in Appendix E.

The highest estimated ground transport maximally exposed worker risk is 0.00052 LCF, just like the marine transport and port worker risks. This estimate is based on the conservative assumption that one truck driver makes enough trips to reach the regulatory limit of 100 mrem per year every year for 13 years. This means that under the assumptions described above, the chance of this individual incurring an LCF due to the basic implementation of Management Alternative 1 would be less than one in a thousand.

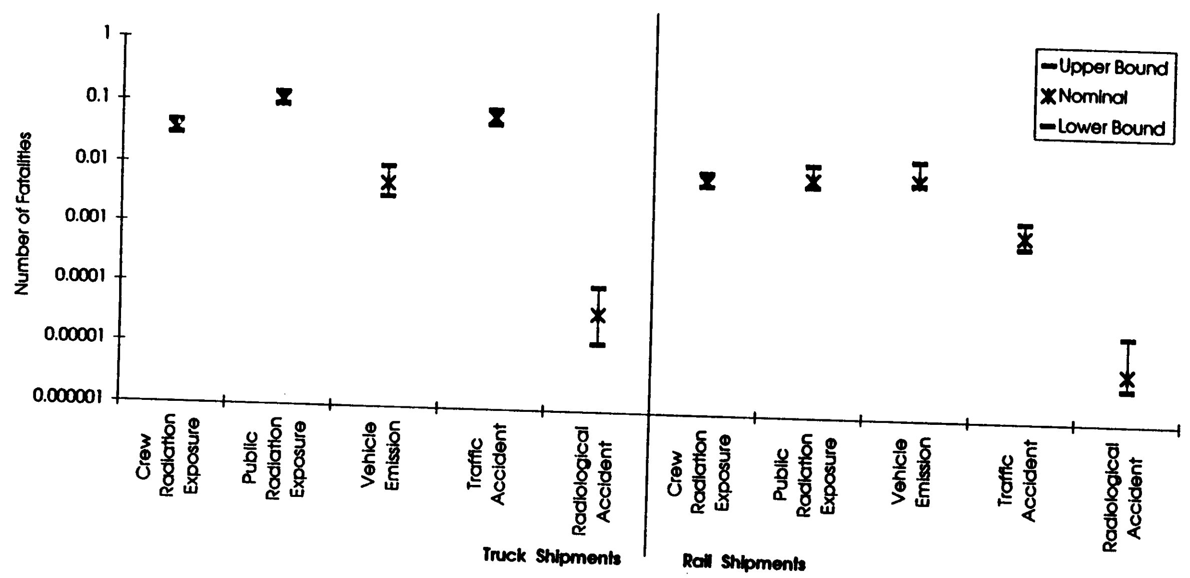
The highest estimated incident-free population risk is 0.30 LCF, which means that there would be a 30 percent chance of one additional cancer fatality among the public and the ground transport workers due to the basic implementation of Management Alternative 1.



- Analysis of Decentralization applies to Programmatic SNF&INEL EIS 1992/1993 Planning Basis and INEL/SRS Regionalization alternatives

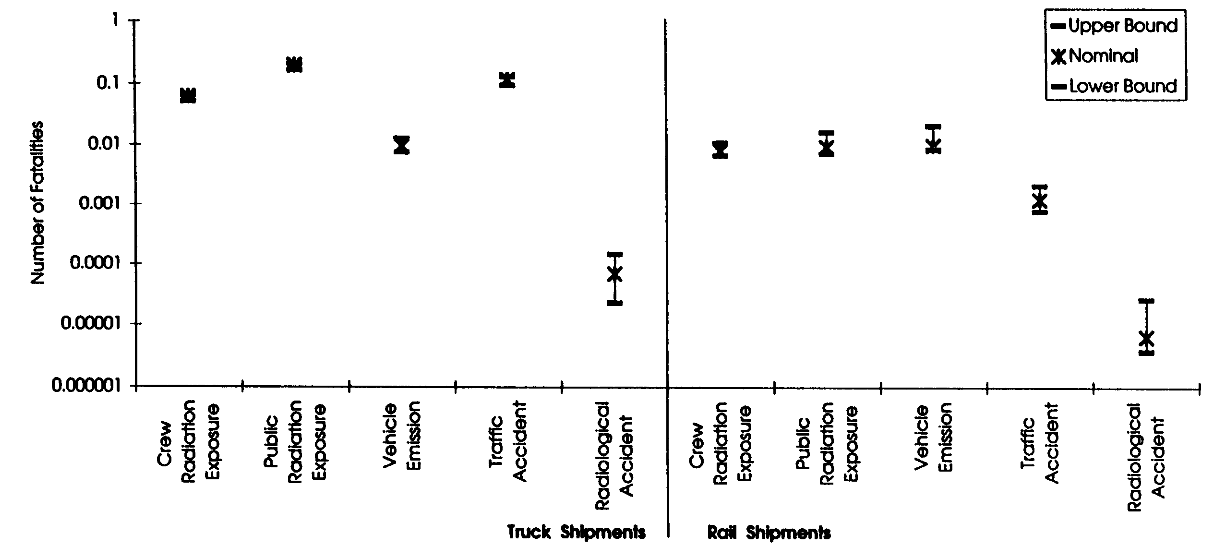
- Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities

Figure 4-2 Range of Estimated Fatalities (Latent and Immediate) Under Basic Implementation of Management Alternative 1 and the Programmatic SNF&INEL Final EIS Decentralization Alternative



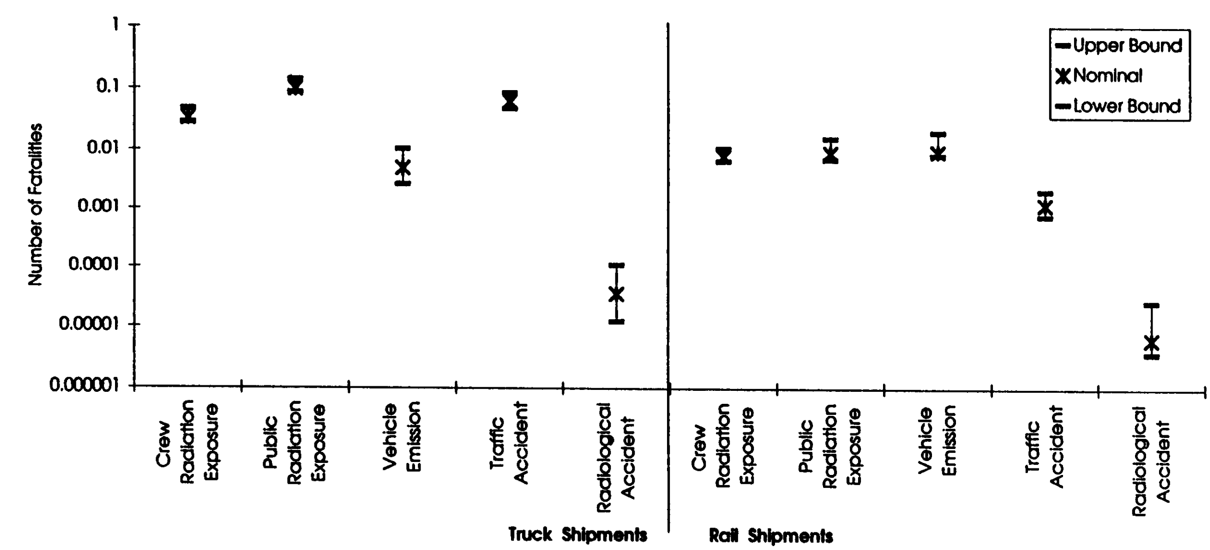
- Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities

Figure 4-3 Range of Estimated Fatalities (Latent and Immediate) Under Basic Implementation of Management Alternative 1 and the Programmatic SNF&INEL Final EIS Regionalization by Fuel Type Alternative



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-4 Range of Estimated Fatalities (Latent and Immediate) Under Basic Implementation of Management Alternative 1 and the Programmatic SNF&INEL Final EIS Centralization to the Savannah River Site Alternative



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-5 Range of Estimated Fatalities (Latent and Immediate) Under Basic Implementation of Management Alternative 1 and the Programmatic SNF&INEL Final EIS Centralization to the Idaho National Engineering Laboratory Alternative

4.2.3.3 Impacts of Accidents During Ground Transport

The most severe accidents that might reasonably occur on this leg of the journey are truck or train crashes, followed by a large fire. If an accident occurred on a causeway at or near a port that caused a cask to fall into seawater, the consequences would be the same as if a cask fell off a ship into seawater. These consequences are presented in Section 4.2.1.3 under the subheading "Sunken Cask." Each State, and most local jurisdictions, maintain a hazardous materials response capability and a radiological protection program. These capabilities, along with the DOE radiological response assets that would be on-call for immediate technical assistance and response, would provide a high-level of expertise and would reduce the potential impacts of a foreign research reactor spent nuclear fuel accident.

Since hazardous materials team training is required to include radiological materials response, each team possesses a basic level of understanding and capability for a foreign research reactor spent nuclear fuel incident response. An incremental enhancement for spent nuclear fuel-specific response characteristics and planning may be required, especially for those jurisdictions along selected routes whose emergency responders are primarily volunteer organizations.

The development of a transportation plan specifically for the shipping campaign that would incorporate and integrate State and local emergency response plans, would increase emergency responder effectiveness and reduce the potential consequences of a foreign research reactor spent nuclear fuel accident.

Each State's emergency planning infrastructure, using the Local Emergency Planning Committees to the State Emergency Response Commission, enables these jurisdictions to identify and resolve potential emergency management and response issues and communicate issues that would require DOE and Department of State attention. This, along with DOE's Transportation External Coordination/Working Group, would ensure that all concerned agencies would be involved in the planning process to address potential problems before they become major hazards.

Risks

The total ground transportation accident risks for the basic implementation of Management Alternative 1 are estimated to range from 0.000004 to 0.00028 LCF from radiation and from 0.001 to 0.14 for traffic fatality, depending on the transportation mode and potential foreign research reactor spent nuclear fuel management sites that might be selected. Section 4.10 compares these risks to those of common activities

The highest estimated MEI radiological risk to members of the public due to accidents during ground transport is 1.4×10^{-11} LCF. This means that the chance of this individual incurring a cancer due to the basic implementation of Management Alternative 1 would be less than one in ten billion.

The highest estimated population radiological risk due to accidents is 0.00028 LCF, which is much less than one LCF.

4.2.3.4 Ground Transport Cumulative Impacts

The Programmatic SNF&INEL Final EIS (DOE, 1995c) analyzed the cumulative impacts of ground transportation, taking into account impacts from: (1) historical shipments of spent nuclear fuel to the five proposed foreign research reactor spent nuclear fuel management sites; (2) the programmatic alternatives; (3) other reasonably foreseeable actions that include transportation of radioactive material; and (4) general radioactive materials transportation that is not related to a particular action. The transportation of foreign research reactor spent nuclear fuel is included in the calculated totals under the spent nuclear fuel shipments for the Programmatic SNF&INEL Final EIS Alternatives 1 through 5. Proposed transportation of all spent nuclear fuel (of which the foreign research reactor fuel is a small component) accounts for less than one percent of the total LCF attributable to the transportation of radioactive material, and foreign research reactor spent nuclear fuel accounts for less than one quarter of that one percent. The total number of LCF over the time period 1943 through 2035 was estimated to be 290.

4.2.3.5 Ground Transport Mitigation Measures

The principal environmental impacts that would occur during ground transport are: (1) LCF due to radiation exposure, (2) LCF due to vehicular emissions, and (3) immediate fatalities due to traffic accidents. All three of these would be reduced by choosing port(s) of entry close to the management site(s). This would minimize the distance that must be covered by the vehicle(s).

Furthermore, in the case of truck transport, the truck driver(s) would be monitored for radiation dose. The annual maximally exposed worker limit of 100 mrem would never be approached during any single shipment, but the same driver could be used for multiple shipments throughout a year. DOE would implement mitigation measures through the foreign research reactor spent nuclear fuel acceptance contracts to ensure that each individual driver's dose remains below the regulatory limit. If any individual truck driver accumulates a dose approaching this limit in a year, DOE would require that new driver(s) be used to keep each individual driver's dose below the regulatory limit.

Since the casks would produce a radiation field of less than 10 mrem/hr at 2 m (6.6 ft) from the vehicle, an individual member of the general public would have to be within 2 m (6.6 ft) of the vehicle for at least ten hours in a year to receive a dose equal to the regulatory limit of 100 mrem/yr. A truck is not likely to sit in a traffic jam right beside another vehicle for as long as ten hours and an individual gas station attendant is not likely to spend ten hours refueling the trucks carrying foreign research reactor spent nuclear fuel. Therefore, DOE does not plan to implement ground transport mitigation measures for members of the general public.

4.2.3.6 Barge Transport

DOE and the Department of State have examined the possibility of using barges for the transport of foreign research reactor spent nuclear fuel as a substitute for truck or rail transport. The only two locations where barge transport is feasible are from the Port of Portland, OR up the Columbia River to the Hanford

Site and from the Port of Savannah, GA up the Savannah River to the Savannah River Site. Barge transport could only be implemented if one or both of these port/site combinations is selected in the Record of Decision.

For barge transport up the Columbia River, the incident-free radiological risk to the public would be approximately 0.0000043 LCF per cask shipment. This is slightly lower than the similar truck and rail shipment risks, which would be 0.000029 and 0.0000058 LCF per shipment, respectively. For barge transport up the Savannah River, the incident-free radiological risk to the public would be approximately 0.0000019 LCF per cask shipment. This is slightly lower than the similar truck and rail shipment risks, which would be 0.000028 and 0.0000026 LCF per shipment, respectively.

For barge transport up the Columbia River, the accident radiological risk due to both airborne and waterborne pathways would be approximately 3.5×10^{-8} LCF per cask shipment. This is slightly higher than the similar truck and rail shipment risks, which would be 1.5×10^{-8} and 3.8×10^{-9} LCF per shipment, respectively. For barge transport up the Savannah River, the accident radiological risk due to both airborne and waterborne pathways would be approximately 2.9×10^{-8} LCF per cask shipment. This is slightly higher than the similar truck and rail shipment risks, which would be 9.4×10^{-9} and 1.1×10^{-9} LCF per shipment, respectively.

The barge transport analysis is presented in more detail in Appendix E, Section E.8.15. The net result is that the foreign research reactor spent nuclear fuel could be transported by barge with approximately the same level of risk to workers and the public as if it was transported by truck or rail.

4.2.3.7 Environmental Justice Along Ground Transport Routes

The dominant radiological risks and impacts associated with incident-free transportation activities are the exposures received by the workers in the immediate vicinity of the casks and people who might be near the casks at truck stops. These individuals would be the only people receiving a measurable exposure during a spent nuclear fuel shipment. As discussed in Section 4.2.3.2, the number of radiation-related latent cancer deaths among transportation workers and the general public combined was calculated to be less than one. The same is true for cancer due to vehicle emissions. Ground transportation accidents would be expected to result in no additional radiological impacts to the population in the vicinity of the accident. Potential impacts from low probability accidents vary considerably and are dependent on the accident conditions (such as the size of the resulting fire, if any) and the weather conditions at the time of an accident. Transportation accidents were estimated to result in no LCF due to radiation and less than 0.2 immediate deaths due to traffic fatalities (see Section 4.2.3.3).

As described in Appendix A, the percentage of the total population comprised of minorities or low-income households varies among routes. Calculations for incident-free and accident conditions demonstrate that for the general population the radiological impacts would be very low. Minority or low-income populations living near these routes would not be subjected to any greater impacts. Therefore, these populations would not receive disproportionately high and adverse impacts. They would be subject to very low impacts as would the general population.

Implementation of the proposed action would have extremely low nonradiological effects on the environment along transportation routes, including the social and economic status of the general population, minority populations, and the low-income population residing along the transportation routes. Economic benefits that would result from increased transportation of cargo along transportation routes would be extremely small for the general population or any particular segment of the population residing along the transportation routes.

4.2.4 Foreign Research Reactor Spent Nuclear Fuel Management Sites

This section presents the potential environmental impacts from the basic implementation of Management Alternative 1 at the potential foreign research reactor spent nuclear fuel management sites, namely the Savannah River Site, the Idaho National Engineering Laboratory, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site. It summarizes the detailed site analysis presented in Appendix F, Sections F.4, F.5, and F.6. The analysis examined environmental topics such as land use, socioeconomics, cultural resources, aesthetic and scenic resources, geology, air quality, water quality, ecology, occupational and public health and safety, noise, traffic and transportation, utilities and energy, and waste management. The analysis showed that the basic implementation of Management Alternative 1 would not have a major effect on any of the environmental topics. Further, none of the environmental topics would clearly differentiate among the potential foreign research reactor spent nuclear fuel management sites.

Because of the public interest in radiation exposure to workers and the public, Section 4.2.4.1 discusses in detail the impacts on occupational and public health and safety from the basic implementation of Management Alternative 1, even though the analysis concludes that such impacts are very low. Section 4.2.4.2 summarizes the impacts on the other environmental topics. Section 4.2.4.3 discusses the cumulative impacts of the basic implementation of Management Alternative 1 at each candidate management site, and Section 4.2.4.4 addresses the waste management and mitigation measures available under the basic implementation of Management Alternative 1. Later in this chapter, Section 4.10 compares the risks of the basic implementation of Management Alternative 1 to risks of

4.2.4.1 Occupational and Public Health and Safety

Possible sources of occupational and public radiological exposure from foreign research reactor spent nuclear fuel include: (1) emissions of radioactive material from incident-free operations, (2) incident-free handling activities, and (3) emissions from accident conditions. Foreign research reactor spent nuclear fuel management is not expected to impact occupational and public health and safety. Nonradiological exposures are not likely to occur during construction or operation of foreign research reactor spent nuclear fuel storage facilities. Radiological exposures are presented in individual subsections for emissions-related impacts, handling-related impacts, and accident-related impacts.

Conservative Assumptions and Impacts to the Public of Incident-Free Site Activities

Doses that could be received by the public during incident-free operation of foreign research reactor spent nuclear fuel storage facilities could only be due to emissions of radioactive material that becomes airborne. The public would be too far from the storage facilities to receive any direct exposure. In summary:

- Doses were calculated for the MEI, defined as an individual living at the management site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the facility. These doses would result from incident-free airborne radiological emissions released during foreign would fail during transport and the associated gaseous fission products would be released during the transfer at the management site. DOE and the Department of State also conservatively assumed that unloading the spent nuclear fuel cask in a dry cell would allow all free gaseous fission products to be released to the environment, while unloading in a wet pool would allow 90 percent of the halogens to be retained in the water. Radiological emissions during wet storage were based on historical data at the Receiving Basin for Offsite Fuels (RBOF) at the Savannah River Site. The emissions during incident-free dry storage would be zero because the spent nuclear fuel would be stored in sealed containers. The methodology and conservative assumptions used for the calculation of radiological emissions under the basic implementation of Management Alternative 1 are discussed in detail in Appendix F, Section F.6.

- Doses were calculated separately for each phase of the program at each candidate management site to accommodate the two-phased implementation of the basic implementation of Management Alternative 1. For example, in the case where the Nevada Test Site, the Hanford Site, or the Oak Ridge Reservation is selected as a Phase 2 site, with the Savannah River Site or the Idaho National Engineering Laboratory as a Phase 1 site, doses were calculated at the Savannah River Site or the Idaho National Engineering Laboratory for Phase 1, and at the Hanford Site, Oak Ridge Reservation, or the Nevada Test Site for Phase 2.

- Doses from an operation which combines an existing wet or dry storage facility for spent nuclear fuel receiving and characterization and dry storage casks to enhance storage capacity are bounded by the doses calculated for the existing facility.

- Doses were conservatively calculated for the maximum quantity of foreign research reactor spent nuclear fuel that could be received at each storage site as discussed in Appendix F, Section F.4.

Tables 4-8 through 4-12 summarize the annual emission-related doses to the public and the associated risks for the MEI and population at each site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period. In general, receipt and unloading at wet storage facilities produces lower public risk than at dry storage facilities.

Table 4-8 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Savannah River Site

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• RBOF (wet storage)	0.00011	5.5×10^{-11}	0.0057	0.0000028
• L-Reactor Basin (wet storage)	0.000073	3.7×10^{-11}	0.0046	0.0000023
• New Dry Storage Facility	0.00018	9.0×10^{-11}	0.0086	0.0000043
Storage at:				
• RBOF (wet storage)	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}
• L-Reactor Basin (wet storage)[a]	0.00036	1.8×10^{-10}	0.022	0.000011
• New Dry Storage Facility	0	0	0	0

[a] *L-Reactor basin doses are due to existing conditions. The foreign research reactor spent nuclear fuel contribution would be six orders of magnitude lower.*

Table 4-9 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Idaho National Engineering Laboratory

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• IFSF[a]/CPP-749 (dry storage)	0.00056	2.8×10^{-10}	0.0045	0.0000023
• Fluorinel Dissolution and Fuel Storage (FAST) (wet storage)	0.00038	1.9×10^{-10}	0.0031	0.0000016
• New Dry Storage Facility[b]	0.00056	2.8×10^{-10}	0.0045	0.0000023
Storage at:				
• IFSF[a]/CPP-749 (dry storage)	0	0	0	0
• FAST (wet storage)	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}
• New Dry Storage Facility[b]	0	0	0	0

[a] *Irradiated Fuel Storage Facility*

[b] *The doses for this new dry storage facility are assumed to be equal to those for IFSF/CPP-749.*

Table 4-10 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Hanford Site

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• Fuel Material Examination Facility (FMEF) (dry storage)	0.00020	1.0×10^{-10}	0.011	0.0000055
• New Dry Storage Facility[a]	0.00025	1.3×10^{-10}	0.015	0.0000075
Storage at:				
• FMEF (dry storage)	0	0	0	0
• New Dry Storage Facility[a]	0	0	0	0

[a] *The doses for this new dry storage facility are different from those for FMEF due to the different release height and location.*

Table 4-11 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Oak Ridge Reservation

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• New Dry Storage Facility	0.089	4.5×10^{-8}	0.085	0.000043
Storage at:				
• New Dry Storage Facility	0	0	0	0

Table 4-12 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Nevada Test Site

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• Engine Maintenance and Disassembly (E-MAD) (dry storage)	0.00076	3.8×10^{-10}	0.00093	4.7×10^{-7}
• New Dry Storage Facility[a]	0.00076	3.8×10^{-10}	0.00093	4.7×10^{-7}
Storage at:				
• E-MAD (dry storage)	0	0	0	0
• New Dry Storage Facility[a]	0	0	0	0

[a] *The doses for this new dry storage facility are assumed to be equal to those for E-MAD.*

Among all the potential foreign research reactor spent nuclear fuel management sites, the maximum estimated annual incident-free public MEI radiological exposure from emissions is 0.09 mrem per year. This exposure would occur at the Oak Ridge Reservation (Table 4-11) during receipt and handling. It is much higher than all other corresponding dose rates in Tables 4-8 through 4-12. The receipt period would be about 3 years, so the total MEI dose would be 0.27 mrem. The associated probability for incurring one LCF would be 1.4×10^{-7} for the MEI, which represents less than two chances in ten million of developing a fatal cancer from radiological exposure.

The highest annual incident-free population risk among the Savannah River Site and the Idaho National Engineering Laboratory (Phase 1 sites) is 0.000011 LCF per year (Tables 4-8 and 4-9), which would be due to emissions from L-Reactor Basin at the Savannah River Site. Assuming some foreign research reactor spent nuclear fuel is stored in this basin for the entire 10 years of Phase 1 plus 3 years to transfer it to a Phase 2 site, the Phase 1 component of this population risk would be as high as 0.00014 LCF. The highest annual incident-free population risk from a new dry storage facility at a potential Phase 2 site (Tables 4-8 through 4-12), is 0.000043 LCF per year, which would be due to receipt/unloading at the Oak Ridge Reservation. Assuming that foreign research reactor spent nuclear fuel could be received at the Oak Ridge Reservation for as long as 3 years, the Phase 2 component of this population risk would be 0.00013 LCF. This is higher than any other combination of Phase 2 dry storage annual risks and durations. Adding the Phase 1 and Phase 2 population risks yields 0.00027 LCF for the total population risk to the public living near the sites due to incident-free conditions.

Conservative Assumptions and Impacts to Workers of Incident-Free Site Activities

Workers would receive radiation doses during handling operations, such as receiving and unloading activities associated with the handling of the cask that transfers the canistered spent fuel to the concrete structure. The worker population doses reported below for new dry storage conservatively reflect the cask design.

- The number of casks handled at each potential foreign research reactor spent nuclear fuel management site would depend on the number of cask shipments considered under the ground transportation options discussed in Section 2.6.4.1, and the amount of foreign research reactor spent nuclear fuel expected to be transferred between facilities during Phase 2.

Table 4-13 provides a summary of the number of casks that would be handled at each potential foreign research reactor spent nuclear fuel management site under the Centralization Alternative in the Programmatic SNF&INEL Final EIS (DOE, 1995c) and in the current EIS.

Table 4-13 Estimated Number of Shipments to and from Each Potential Foreign Research Reactor Spent Nuclear Fuel Management Site

Candidate Storage Site	*Incoming Shipments*	*Intersite Shipments*	*Outgoing Shipments*	*Total Shipments*
Savannah River Site or Idaho National Engineering Laboratory Phase 1	644[a]	0	161	805
Savannah River Site or Idaho National Engineering Laboratory Phases 1 and 2	837[b]	209	0	1,046
Hanford Site or Oak Ridge Reservation or Nevada Test Site Phase 2	354[c]	0	0	354

[a] *10-year receipt in foreign research reactor spent nuclear fuel certified casks.*

[b] *13-year receipt in foreign research reactor spent nuclear fuel certified casks.*

[c] *161 from near term site using large truck casks and 193 from ports using foreign research reactor spent nuclear fuel certified casks.*

Tables 4-14 through 4-18 present the population doses and risks that would be received by the members of the working crew, if that crew handled the total number of casks at each management site. The results do not include shipments in large rail casks.

Table 4-14 Handling-Related Impacts to Workers at the Savannah River Site

	Worker Population Dose (person-rem)	*Worker Population Risk (LCF)*

Table 4-16 Handling-Related Impacts to Workers at the Hanford Site

	Worker Population Dose (Person-rem)	Worker Population Risk (LCF)
	FMEF/New Dry Storage	FMEF/New Dry Storage
Phase 2	266[a]	0.11[a]

[a] *Cask design*

Table 4-17 Handling-Related Impacts to Workers at the Oak Ridge Reservation

	Worker Population Dose (Person-rem)	Worker Population Risk (LCF)
	New Dry Storage	New Dry Storage
Phase 2	266[a]	0.11[a]

[a] *Cask design*

Table 4-18 Handling-Related Impacts to Workers at the Nevada Test Site

	Worker Population Dose (person-rem)		Worker Population Risk (LCF)	
	E-MAD	New Dry Storage	E-MAD	New Dry Storage
Phase 2	113	266[a]	0.05	0.11[a]

[a] *Cask design*

According to the above tables, the highest dose to a working crew at a single site would be 424 person-rem at the Idaho National Engineering Laboratory in the analyzed case which assumes that all foreign research reactor spent nuclear fuel is received in the Irradiated Fuel Storage Facility and/or the CPP-749 facility (dry storage) during Phase 1 and is transferred to a new dry storage facility at the Idaho National Engineering Laboratory in Phase 2. The associated number of additional LCF is 0.17. The highest dose to working crews for both phases in more than one site is 523 person-rem: 266 person-rem at one of the 3 Phase 2 sites, plus 257 person-rem at the Idaho National Engineering Laboratory as the Phase 1 site. The associated probability for developing one LCF among the working crews of the two sites is 0.21.

Conservative Assumptions and Accident-Related Impacts

An evaluation of hypothetical accidental radioactive material releases at the potential foreign research reactor spent nuclear fuel management sites was performed to assess the impact of possible radiation exposure to individuals and the general population (see also Appendix F, Section F.6). All inputs are site-specific except for the radioactivity release. Site-specific information includes meteorological conditions, population distribution, and food production and consumption rates within 80 km (50 mi) of the management location.

The radiation doses to the following individuals and the general population are calculated for accident conditions at the spent nuclear fuel management facility:

- Worker: An individual located 100 m (330 ft) from the radioactive material release point. (The impact of accidents on close-in workers is not calculated numerically but is discussed qualitatively for each accident at the end of this section.) For elevated release (from a tall stack), the worker dose was calculated at a point of maximum dose. The distance at which the maximum dose occurs is frequently greater than 100 m (330 ft) for elevated release.

The direction to the worker was chosen as the direction to the maximally exposed sector. The dose to the worker is calculated for the 50th-percentile meteorological condition (DOE, 1992a).

- Maximally Exposed Individual (MEI): A theoretical member of the general public living at the management site boundary receiving the maximum exposure. This individual is conservatively assumed to be located in a direction downwind from the release point. The dose to the MEI is shown for the conservative 95th-percentile meteorological condition.

- Nearest Public Access Individual (NPAI): An individual stranded on a highway or public access road near to the facility at the time of an accident. The distance to the NPAI was chosen as the distance to the nearest public access point; the direction was chosen as the direction to that point. The dose to the NPAI is shown for the conservative 95th-percentile meteorological condition.

- General population within an 80-km (50-mi) radius of the facility: The dose calculations are performed for the direction downwind from the release point that results in highest dose to the public. The dose to the population is shown for the conservative 95th-percentile meteorological condition.

The radiation dose to individuals and the public resulting from exposure to radioactive contamination was calculated using external (direct exposure), inhalation, and ingestion pathways. Dispersion in air from point of release was estimated with both 50th-percentile and 95th-percentile meteorological conditions. The 50th-percentile condition represents the median meteorological condition. The 95th-percentile condition is defined as that condition which is not exceeded more than 5 percent of the time, and is more conservative than the 50th-percentile condition.

The ingestion dose is calculated by considering that the individual and the public would consume the contaminated food produced in the vicinity [up to 80 km (50 mi)] of the accident. This is conservative, and it is expected that continued consumption of contaminated food products by the public would be suspended if the projected dose exceeded the protective action guidelines developed by the U.S. Environmental Protection Agency (EPA, 1991a). To ensure a consistent and conservative analytical basis, no reduction of exposure due to a protective action guideline was used in this analysis.

Accidents considered for detailed analysis are similar to those that were analyzed in the Programmatic SNF&INEL Final EIS. The selection of the accidents was based on the following considerations:

(1) criticality caused by human error during operation, equipment failure, or earthquake; (2) mechanical damage to foreign research reactor spent nuclear fuel during examination and preparation (cropping off the aluminum and nonfuel end of a spent fuel element); and (3) accident involving an impact by either an internal or an external initiator with and without an ensuing fire.

Six accident scenarios were evaluated at each management location using identical source terms (estimated amounts of radioactive material released during postulated accidents). The wet pool accidents are assumed to be cutting into the fuel region or mechanical damage due to operator error, an accidental criticality, and an aircraft crash into the water pool facility. The dry storage accidents are assumed to be cutting into the fuel region or mechanical damage during examination work and handling in a dry cell, dropping of a spent nuclear fuel cask, and an aircraft crash with an ensuing fire.

Tables 4-19 through 4-23 present the frequencies and the consequences of postulated accidents to the offsite MEI, NPAI, and offsite population for the 95th-percentile meteorological conditions using the conservative assumptions and input values discussed above. The worker doses are calculated only for the

accident. DOE and the Department of State did not estimate the worker population dose due to accidents.

Table 4-19 Frequency and Consequences of Accidents at the Savannah River Site

	Frequency (per yr)	Consequences							
		MEI		NPAI		Population		Worker	
		(mrem)	(LCF)	(mrem)	(LCF)	(person-rem)	(LCF)	(mrem)	(LCF)
Dry Storage Accidents[a]									
Spent Nuclear Fuel									

Table 4-21 Frequency and Consequences of Accidents at the Hanford Site

	Frequency (per yr)	Consequences							
		MEI		NPAI		Population		Worker	
		(mrem)	(LCF)	(mrem)	(LCF)	(person-rem)	(LCF)	(mrem)	(LCF)
Dry Storage Accidents[a]									
Spent Nuclear Fuel Assembly Breach	0.16	3.0	0.0000015	0.57	2.9×10^{-7}	42	0.021	50	0.000020
Dropped Spent Nuclear Fuel Cask	0.0001	0.26	1.3×10^{-7}	0.0085	4.3×10^{-9}	3.0	0.0015	0.22	8.8×10^{-8}
Aircraft Crash w/Fire[b]	NA	NA	NA	NA	NA	NA	NA	NA	NA
Dry Storage Accidents at FMEF									
Spent Nuclear Fuel Assembly Breach[c]	0.16	4.7	0.0000024	2.1	0.0000011	46	0.023	0.99	4.0×10^{-7}
Dropped Spent Nuclear Fuel Cask[c]	0.0001	0.2	1.0×10^{-7}	0.032	1.6×10^{-8}	3.2	0.0016	0.0049	2.0×10^{-9}
Aircraft Crash w/Fire[b]	NA	NA	NA	NA	NA	NA	NA	NA	NA

[a] *New Dry Storage Facility*

[b] *Aircraft Crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

[c] *Emissions would be released through a tall stack, so workers would receive low doses.*

NA = Not applicable

Table 4-22 Frequency and Consequences of Accidents at the Oak Ridge Reservation

	Frequency (per yr)	Consequences							
		MEI		NPAI		Population		Worker	
						(person-			

The analyses were performed for a generic dry storage at the five potential foreign research reactor spent nuclear fuel management sites, as well as for site-specific locations (i.e., FMEF at the Hanford Site, E-MAD at the Nevada Test Site, L-Reactor Basin and RBOF at the Savannah River Site).

Multiplying the frequency of each accident times its consequences at each site and converting the radiation doses to LCF yields the annual risks associated with each potential accident at each candidate management site. These annual risk estimates are presented in Tables 4-24 through 4-28.

Table 4-24 Annual Risks of Accidents at the Savannah River Site

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)

Table 4-26 Annual Risks of Accidents at the Hanford Site

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
Dry Storage Accidents[a]				
Spent Nuclear Fuel Assembly Breach	2.4×10^{-7}	4.6×10^{-8}	0.0034	0.0000032
Dropped Spent Nuclear Fuel Cask	1.3×10^{-11}	4.3×10^{-13}	1.5×10^{-7}	8.8×10^{-12}
Aircraft Crash w\Fire[b]	NA	NA	NA	NA
Dry Storage Accidents at FMEF				
Spent Nuclear Fuel Assembly Breach[c]	3.7×10^{-7}	1.7×10^{-7}	0.0037	6.4×10^{-8}
Dropped Spent Nuclear Fuel Cask[c]	8.0×10^{-12}	1.6×10^{-12}	1.6×10^{-7}	2.5×10^{-13}
Aircraft Crash with Fire[b]	NA	NA	NA	NA

[a] *New Dry Storage Facility*

[b] *Aircraft crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

[c] *Emissions would be released through a tall stack.*

NA = Not applicable

Table 4-27 Annual Risks of Accidents at the Oak Ridge Reservation

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
Dry Storage Accidents[a]				
Spent Nuclear Fuel Assembly Breach	0.0000018	0.0000034	0.0044	0.0000088
Dropped Spent Nuclear Fuel Cask	7.0×10^{-11}	9.0×10^{-12}	7.5×10^{-7}	2.4×10^{-11}
Aircraft Crash w\Fire	1.2×10^{-9}	9.0×10^{-11}	0.0000015	2.4×10^{-10}

[a] *New Dry Storage Facility*

Table 4-28 Annual Risks of Accidents at the Nevada Test Site

	Risks			

component of this population risk would be 0.013 LCF. This is higher than any other combination of Phase 2 annual accident risks and associated durations. Adding the Phase 1 and Phase 2 population risks yields 0.11 LCF for the total population risk due to accidents.

Impacts of Accidents on Close-in Workers

Secondary Impacts of Accidents

Impacts of accidents on resources other than human health and safety (secondary impacts), have been addressed in Section F.4 for each management site. The general conclusion is that no measurable secondary impacts to land uses, cultural resources, water quality, ecological resources, national defense, or local economies are expected from the postulated accidents involving foreign research reactor spent nuclear fuel at the management sites.

4.2.4.2 Topics Not Discussed in Detail

This section summarizes the potential impacts for the environmental topics not covered in Section 4.2.4.1, namely land use, socioeconomics, cultural resources, aesthetic and scenic resources, geology, air quality, water quality, ecology, noise, utilities and energy, and waste management. The detailed analysis of these topics presented in Appendix F, Section F.4 showed that none of these topics clearly differentiated among the potential foreign research reactor spent nuclear fuel management sites nor had any major environmental impact. The discussion of each topic generally concentrates on management sites and alternatives that have the largest estimated impacts, and demonstrates that the environmental impacts for that topic are not of sufficient magnitude to be given strong consideration in the decision making process.

4.2.4.2.1 Land Use

The basic implementation of Management Alternative 1 would only result in minor land use impacts at any of the potential foreign research reactor spent nuclear fuel management sites. The largest land use impact would be 16 ha (40 acres) at the Oak Ridge Reservation to construct a new dry storage facility. This represents less than 0.1 percent of the total size of the Oak Ridge Reservation. A description of the land use impacts at the other potential foreign research reactor spent nuclear fuel management sites is contained in Appendix F.4. For all of the potential foreign research reactor spent nuclear fuel management sites, new foreign research reactor spent nuclear fuel storage facilities would be built on land previously disturbed or designated for industrial use. No additional land outside of the existing sites would be required for foreign research reactor spent nuclear fuel management. It should be noted that land use and other environmental impacts associated with the construction activities would be minimal, under the implementation alternatives that use refurbishment of existing facilities for interim storage (i.e., BNFP at the Savannah River Site and E-MAD at the Nevada Testing Site). All environmental impacts from the refurbishment and operation of these facilities would be bounded by the impacts associated with the construction and operation of new generic storage facilities. Land use impacts are discussed in more detail in Appendix F, Section F.4.

4.2.4.2.2 Socioeconomics

The basic implementation of Management Alternative 1 would only result in minor socioeconomic impacts at any of the potential foreign research reactor spent nuclear fuel management sites. Socioeconomic impacts are defined for purposes of this analysis in terms of direct effects, which include changes in site employment and expenditures from foreign research reactor spent nuclear fuel-related construction and operation and indirect effects, such as changes that result from regional purchases, nonpayroll expenditures, and payroll spending by site employees.

No construction personnel would be needed for existing facilities, and not more than 240 workers per year (peak) would be needed to build a new dry storage facility. The annual staffing requirements for operations would be about 30 and 8 full-time employees during receipt and storage, respectively, for a new dry storage facility. This would represent 0.15 to 0.9 percent of the existing work force at any of the potential foreign research reactor spent nuclear fuel management sites. No new hiring would be expected because most positions would be filled by reassignments of the existing work force. Even if all operational positions were filled by new hires, this would represent about an even smaller increase in regional employment. The secondary effects would be even lower.

4.2.4.2.3 Cultural Resources

The basic implementation of Management Alternative 1 would only result in minor cultural impacts at any of the potential foreign research reactor spent nuclear fuel management sites. Cultural, archaeological, historic, and architectural resources are defined as prehistoric and historic sites, districts, structures, and evidence of human use that are considered to be important to a culture, subculture, or a community for scientific, traditional, religious, or other reasons.

Although most of the potential foreign research reactor spent nuclear fuel management sites contain areas of archaeological, cultural, or historical interest, little or no direct impacts on cultural resources would be expected because of the location of the foreign research reactor spent nuclear fuel storage facilities. Specific site surveys have not been completed; however, based on existing information, no known cultural resources would be affected by construction or operation of foreign research reactor spent nuclear fuel facilities. Prior to construction, specific site surveys would be conducted. In the event that cultural resources were encountered during construction, the State Historic Preservation Officer would be contacted immediately. Similarly, Tribal leaders would be notified if any Native American resources were found.

4.2.4.2.4 Aesthetic and Scenic Resources

The basic implementation of Management Alternative 1 would only result in minor impacts to aesthetic and scenic resources at any of the potential foreign research reactor spent nuclear fuel management sites. Foreign research reactor spent nuclear fuel storage facilities would be located far from public view in areas previously disturbed or designated for industrial use. Construction activities would generate fugitive dust that could temporarily affect visibility. However, best management practices would be implemented to minimize such conditions. Furthermore, facility operations would not produce emissions that would adversely impact visibility.

4.2.4.2.5 Geology

The basic implementation of Management Alternative 1 would only result in minor geologic impacts at any of the potential foreign research reactor spent nuclear fuel management sites. Except for the potential existence of gold, tungsten, and molybdenum at the Nevada Test Site, geologic resources consist only of surficial sand, gravel, or clay deposits, all of which have low economic value. Construction activities would disturb these surface deposits, but because of the large volume of these materials on the potential foreign research reactor spent nuclear fuel management sites, the impact would be expected to be small.

4.2.4.2.6 Air Quality

The basic implementation of Management Alternative 1 would only result in minor impacts on air quality at any of the potential foreign research reactor spent nuclear fuel management sites. The projected emissions from foreign research reactor spent nuclear fuel storage at the potential management sites would not contribute to Federal or State nonattainment standards. Construction activities would be expected to cause only temporary, minor increases in fugitive dust emissions, but the use of standard dust suppression techniques would be expected to mitigate this problem. Particulate emissions could temporarily affect visibility in localized areas, but would not adversely affect Federal or State attainment standards.

4.2.4.2.7 Water Quality

The basic implementation of Management Alternative 1 would have only minor impacts on water resources at the potential foreign research reactor spent nuclear fuel management sites. Water consumption during construction would require very small amounts of water when compared to daily water usage at the potential management sites.

During operations, the greatest amount of water consumed annually would be about 2.1 million liters (550,000 gal) per year. This amount represents no more than 0.2 percent of the annual water consumption at any of the potential foreign research reactor spent nuclear fuel management sites. At the Nevada Test Site, where available water is limited, a cumulative water supply impact could be important from activities other than foreign research reactor spent nuclear fuel management, but the foreign research reactor spent nuclear fuel management contribution would be very small. Further study of the Ash Meadows sub-basin would be required to specify the exact impact on aquifer yield and integrity.

Under normal operations there would be no direct discharge of effluent to ground or surface waters from a new dry storage facility.

4.2.4.2.8 Ecology

The basic implementation of Management Alternative 1 would only result in minor ecological impacts at the potential foreign research reactor spent nuclear fuel management sites. Under any construction of new facilities, individual or small populations of some wildlife species could be disturbed, displaced, or destroyed. However, the size of the areas affected would be small in relation to the size of the potential foreign research reactor spent nuclear fuel management sites and the size of remaining natural habitats. The type of habitats affected could vary but would be typical of the regional area in which the foreign research reactor spent nuclear fuel storage facility is located. For this reason, any such habitat losses would probably not affect any threatened or endangered species or critical habitats in the area. Habitat fragmentation is not expected because new storage facilities would be constructed on land that has been previously disturbed or designated for industrial purposes. Mitigation plans would be developed in consultation with the appropriate agencies if any threatened or endangered species were identified.

DOE has begun or has completed consultations with the U.S. Fish and Wildlife Service regarding threatened and endangered species for the proposed construction site of foreign research reactor spent nuclear fuel storage facilities at the five potential sites, as required by the Endangered Species Act.

4.2.4.2.9 Noise

The basic implementation of Management Alternative 1 would only result in minor noise impacts at the potential foreign research reactor spent nuclear fuel management sites. Construction activities would generate noise levels consistent with light industrial activity. Based on existing studies these noises would not be expected to propagate offsite at levels that would affect the general population. Noises generated during operations would be less than those during construction.

4.2.4.2.10 Materials, Utilities, and Energy

The basic implementation of Management Alternative 1 would only result in minor impacts on materials, utilities, and energy at the potential foreign research reactor spent nuclear fuel management sites. For existing facilities, incremental increases in materials, utilities, and energy would be very small. New dry storage facilities would result in increased demands on water, power, and sewage. The increased water usage during construction would add no more than 0.2 percent to existing sitewide levels. Increased annual electricity requirements would be about 800 to 1,000 megawatt hours per year and the increased sewage generation would be no more than 1.59 million liters per year (420,000 gal per year), which is less than one percent above existing sitewide levels. At the Nevada Test Site, a central sewage treatment system would have to be constructed for spent nuclear fuel management activities, which would include the foreign research reactor spent nuclear fuel storage facilities. However, all other existing system capacities could manage the estimated increases for materials, utilities, and energy.

4.2.4.2.11 Waste Management

The basic implementation of Management Alternative 1 would only result in minor waste management impacts at the potential foreign research reactor spent nuclear fuel management sites. At all potential management sites the amount of waste generated from foreign research reactor spent nuclear fuel storage is very small when compared to the annual waste projection for each site.

4.2.4.3 Key Cumulative Impacts at the Potential Foreign Research Reactor Spent Nuclear Fuel Management Sites

All of the potential foreign research reactor spent nuclear fuel management sites contain facilities unrelated to foreign research reactor spent nuclear fuel that may continue to operate throughout the foreign research reactor spent nuclear fuel program (approximately 40 years). Impacts from both construction and operation of foreign research reactor spent nuclear fuel facilities would be cumulative with the impacts of existing and planned facilities or actions such as environmental restoration and waste management activities unrelated to foreign research reactor spent nuclear fuel and impacts from the management of DOE's spent nuclear fuel inventory.

This section compares the impacts of the basic implementation of Management Alternative 1 and of the implementation alternatives presented in Section 4.3 to the cumulative impacts at each site. The site-specific cumulative impacts are discussed in more detail in Appendix F.

4.2.4.3.1 Key Cumulative Impacts at the Savannah River Site

Table 4-29 presents the key cumulative impacts from ongoing actions and reasonably foreseeable actions at the Savannah River Site, including:

Table 4-29 Key Cumulative Impacts at the Savannah River Site

Environmental Impact Parameter	FRR SNF Receipt and Storage Contribution	FRR SNF Receipt and Chemical Separation Contribution	Current Activities[a]	Other Activities[b]	Cumulative Impact
Occupational and Public Health and Safety:					
• MEI Dose (mrem/yr)	0.00036	0.66	0.25	4.1	5.0
LCF (per year)	1.8×10^{-10}	3.3×10^{-7}	1.25×10^{-7}	0.000002	0.0000025
• Population Dose (person-rem/yr)	0.022	27	9.1	295	331
LCF (per year)	0.000011	0.014	0.0045	0.15	0.17
• Worker Collective Dose (person-rem/yr)	10[c]	21	263	1,418	1700
LCF (per year)	0.004	0.0084	0.10	0.57	0.68
Waste Generation:					
• High-Level (canisters/yr)	0	6.5	(d)	190[e]	190[e]
• Saltstone (m^3/yr)	0	370	(d)	60,000	60,000
• Transuranic (m^3/yr)	0	0	(d)	1,038	1,038
• Mixed/Hazardous (m^3/yr)	0	8	(d)	2,561	2,569
• Low-Level (m^3/yr)	22	5,700	(d)	35,600	41,300

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

[a] *Based on 1993 site data*

[b] *Other activities include: interim management of nuclear materials, spent nuclear fuel management, Vogtle plant operation, defense waste processing facility, stabilization of plutonium-solutions, site-wide waste management activities, tritium accelerator facility, disposition of surplus HEU, storage and disposition of weapons-usable fissile materials, and the stockpile stewardship and management program activities.*

[c] *The dose is due to the handling of the FRR SNF during receipt and transfer between facility, averaged over 40 years.*

[d] *Included in "other activities"*

[e] *Expected Defense Waste Processing Facility canister production rate (DOE, 1995b).*

- The operation of the Vogtle Electric Generating Plant located approximately 16 km (10 mi) south west of the center of the Savannah River Site.

- The implementation of the preferred alternative in the Management of Nuclear Materials EIS.

- Shipment of aluminum-based spent nuclear fuel to the Savannah River Site for storage and disposal discussed in Appendix C of the Programmatic SNF & INEL Final EIS.

- Stockpile Stewardship and Management Program.
- Current Savannah River Site projects (based on 1993 data).

Table 4-29 also shows the impacts of receipt and near-term chemical separation at the Savannah River Site, from Implementation Alternative 6 of Management Alternative 1 in Section 4.3.6. These impacts are sufficiently distinct from those of the other alternatives that they are presented separately. These impacts would occur only while the chemical separation facilities are operating.

The results in Table 4-29 show that the contribution of foreign research reactor spent nuclear fuel to the cumulative impacts at the Savannah River Site would be minimal.

4.2.4.3.2 Key Cumulative Impacts at the Idaho National Engineering Laboratory

Table 4-30 presents the key cumulative impacts from ongoing actions and reasonably foreseeable actions at the Idaho National Engineering Laboratory, including the proposed construction and operation of an accelerator facility for tritium production (along with associated support facilities), the management of DOE-owned spent nuclear fuel discussed in Appendix B of the Programmatic SNF&INEL Final EIS, and the storage and disposition of weapons-usable fissile materials at the Idaho National Engineering Laboratory site.

Table 4-30 Key Cumulative Impacts at the Idaho National Engineering Laboratory

Environmental Impact Parameter	FRR SNF Receipt and Storage Contribution	FRR SNF Receipt and Chemical Separation Contribution	Current Activities[a]	Other Activities[a]	Cumulative Impact
Occupational and Public Health and Safety:					
• MEI Dose (mrem/yr)	0.00056	0.048	0.056	0.0057	0.11
LCF (per year)	2.8×10^{-10}	2.4×10^{-8}	2.8×10^{-8}	2.8×10^{-9}	5.5×10^{-8}
• Population Dose (person-rem/yr)	0.0045	0.39	0.34	32	33
LCF (per year)	2.3×10^{-6}	0.00020	0.00017	0.016	0.016
• Worker Collective Dose (person-rem/yr)	10[b]	18	30	344	392
LCF (per year)	0.004	0.0072	0.012	0.137	0.16
Waste Generation:					
• High-Level (canisters/yr)	0	7.5	0	327[c]	327[c]
• Grout (m^3/yr)	0	167	0	875[d]	875[d]
• Transuranic (m^3/yr)	0	0	712	46	758
• Mixed/Hazardous (m^3/yr)	0	8	243	8	259
• Low-Level (m^3/yr)	22	5,700	4,795	2,800	13,300

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

[a] *Other activities include: DOE-owned spent nuclear fuel management, construction and operation of a tritium accelerator facility, and the disposition of weapons-usable fissile materials.*

[b] *The dose is due to the handling of FRR SNF during receipt and transfer, averaged over 40 years.*

[c] *Assumed canister production rate (DOE, 1995b).*

[d] *Design capacity of the proposed Waste Immobilization Facility, which is not funded.*

Table 4-30 also shows the impacts of receipt and near-term chemical separation at the Idaho National Engineering Laboratory, from Implementation Alternative 6 of Management Alternative 1 in Section 4.3.6. These impacts are sufficiently distinct from those of the other alternatives that they are presented separately. These impacts would occur only while the chemical separation facilities are operating.

The results in Table 4-30 show that the contribution of foreign research reactor spent nuclear fuel management to the cumulative impacts at the Idaho National Engineering Laboratory would be minimal.

4.2.4.3.3 Key Cumulative Impacts at the Hanford Site

Table 4-31 presents the key cumulative impacts from ongoing actions and reasonably foreseeable actions at the Hanford Site, including those discussed in the Programmatic SNF&INEL Final EIS, the Management of Spent Nuclear Fuel from the K Basins Draft EIS, and the Safe Interim Storage of Hanford Tank Wastes Final EIS.

Table 4-31 Key Cumulative Impacts at the Hanford Site

Environmental Impact Parameter	*FRR SNF Contribution*	*Other Activities*[a]	*Cumulative Impact*
Occupational and Public Health and Safety:			
• MEI Dose (mrem/yr)	0.00025	0.0036	0.0036
LCF (per year)	1.3×10^{-10}	1.5×10^{-9}	1.5×10^{-9}
• Population Dose (person-rem/yr)	0.015	0.22	0.235
LCF (per year)	0.0000075	0.00011	0.00011
• Worker Collective Dose (person-rem/yr)	8.9[b]	116.5	125.4
LCF (per year)	0.0035	0.0466	0.05
Waste Generation:			
• High-Level (canisters/yr)	0	320[c]	320[c]
• Transuranic (m^3/yr)	0	240	240
• Mixed/Hazardous (m^3/yr)	0	402	402
• Low-Level (m^3/yr)	22	33,310	33,332

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

a [illegible]

Table 4-32 Key Cumulative Impacts at the Oak Ridge Reservation

Environmental Impact Parameter	FRR SNF Contribution	Other Activities[a]	Cumulative Impact
Occupational and Public Health and Safety:			
• MEI Dose (mrem/yr)	0.09	15.5	15.6
LCF (per year)	4.5×10^{-8}	0.0000077	0.0000078
• Population Dose (person-rem/yr)	0.085	94.5	94.6
LCF (per year)	0.000043	0.047	0.047
• Worker Collective Dose (person-rem/yr)	8.9[b]	261.3	270.2
LCF (per year)	0.0036	0.104	0.108
Waste Generation:			
• High-Level (canisters/yr)	0	0	0
• Transuranic (m^3/yr)	0	16	16
• Mixed/Hazardous (m^3/yr)	0	119,411	119,411
• Low-Level (m^3/yr)	22	34,989	35,011

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

[a] *Other activities include: DOE-owned spent nuclear fuel management, construction and operation of the Expended Core Facility, the construction and operation of the Advanced Neutron Source Facility, construction and operation of a Tritium production facility, and surplus highly-enriched uranium management activities at the site.*

[b] *The dose is due to the handling of FRR SNF during receipt, averaged over 30 years.*

The results in Table 4-32 show that the contribution from storage of foreign research reactor spent nuclear fuel to the cumulative impacts at the Oak Ridge Reservation would be minimal.

4.2.4.3.5 Key Cumulative Impacts at the Nevada Test Site

Table 4-33 presents the key cumulative impacts from ongoing actions and reasonably foreseeable actions at the Nevada Test Site, including those discussed in the Programmatic SNF&INEL Final EIS and the Tritium Supply and Recycling Final EIS. The Programmatic SNF&INEL Final EIS includes the quantitative impacts from a proposed Expended Core Facility at the Site. The Nevada Test Site is also considered in the storage and disposition of weapons-usable fissile materials program which could affect the site environment. The impacts from this program have not been determined sufficiently at this time to allow impact evaluation.

The results in Table 4-33 show that the contribution from storage of foreign research reactor spent nuclear fuel to the cumulative impacts at the Nevada Test Site would be minimal.

4.2.4.4 Waste Minimization and Mitigation Measures at the Potential Foreign Research Reactor Spent Nuclear Fuel Management Sites

Although environmental impacts at the potential foreign research reactor spent nuclear fuel management sites would be minimal in all environmental media and mitigation measures would not be necessary, the sites would implement measures in some areas to minimize impacts. Mitigation measures would be taken in the areas of pollution control, socioeconomics, cultural resources, air and water resources, occupational and public health and safety, and accident prevention. Appendix F provides details on these issues.

Table 4-33 Key Cumulative Impacts at the Nevada Test Site

Environmental Impact Parameter	FRR SNF Contribution	Other Activities[a]	Cumulative Impact
Occupational and Public Health and Safety:			
• MEI Dose (mrem/yr)	0.00076	0.31	0.31
LCF (per year)	3.8×10^{-10}	1.55×10^{-7}	1.55×10^{-7}
• Population Dose (person-rem/yr)	0.00093	0.095	0.095
LCF (per year)	4.7×10^{-7}	0.00047	0.000047
• Worker Collective Dose (person-rem/yr)	8.9[b]	81	89.9
LCF (per year)	0.0036	0.032	0.035
Waste Generation:			
• High-Level (canisters/yr)	0	0	0
• Transuranic (m^3/yr)	0	16	16
• Mixed/Hazardous (m^3/yr)	0	252	252
• Low-Level (m^3/yr)	22	44,578	44,600

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

[a] *Other activities include existing activities, DOE-owned spent fuel management activities, construction and operation of an Expended Core Facility, and construction and operation of a tritium production facility.*

[b] *The dose is due to the handling of foreign research reactor spent nuclear fuel during receipt, averaged over 30 years.*

4.2.4.5 Environmental Justice at the Potential Foreign Research Reactor Spent Nuclear Fuel Management Sites

Under incident-free foreign research reactor spent nuclear fuel management site activities associated with receipt and storage of the spent nuclear fuel, the dominant radiological impacts would be the exposures received by the site workers in the immediate vicinity of the spent nuclear fuel container. These individuals are principally those working within the spent nuclear fuel storage facility. As discussed in Section 4.2.4.1, under incident-free operating conditions, no radiological fatalities would be expected among radiation workers or the general public.

Section 4.2.4.1 also discusses radiological effects due to accidents for both wet storage and dry storage. As shown in Tables 4-24 through 4-28, the dominant radiological risks due to accidents are estimated to occur during breach of a spent nuclear fuel assembly. No LCF are expected to result from the basic implementation of Management Alternative 1.

Appendix A describes minority populations and low-income households residing near candidate management sites. Table 4-34 summarizes this description. Calculations for incident-free and accident conditions demonstrate that for the general population the impacts would be very low. Minority or low-income populations living near the potential management sites would not be subjected to any greater impacts. Therefore, these populations would not receive disproportionately high and adverse impacts. They would be subject to very low impacts, as would the general population.

Table 4-34 Summary Description of Minority Populations and Low-Income Households Residing Within 80 km (50 mi) of Candidate Management Sites

Candidate Management Site	Total Population	Minority Population	Total Households	Low-Income Households
Savannah River Site	566,823	214,016	197,937	82,930
Idaho National Engineering Laboratory	176,311	15,449	55,109	22,452
Hanford Site	383,934	95,042	136,496	57,667
Oak Ridge Reservation	863,758	53,185	335,589	147,537
Nevada Test Site	12,421	2,005	4,194	2,024

Characterization of the number and location of minority and low-income populations is dependent on how these populations are defined and what assumptions are used in conducting the analysis. As discussed in Appendix A, at the time this Final EIS and the Programmatic SNF&INEL Final EIS were prepared, the Federal Interagency Working Group on Environmental Justice had not issued final guidance on the definitions of minority and low-income populations, or the approach to be used in analyzing environmental justice, as directed by the Executive Order. Final internal DOE guidance on environmental justice has also not been adopted. As a result, both the definitions and assumptions used by and within agencies for conducting environmental justice analyses can vary, and the resulting demographic results can differ on a case-by-case basis. For example, this Final EIS and the Programmatic SNF&INEL Final EIS present demographic characterizations derived from the same United States Census Bureau data base, but these documents used different definitions and assumptions. Several of the same candidate interim spent nuclear fuel management sites were evaluated in both documents. As discussed in Appendix A, variations in these definitions and assumptions led to differences in the characterization of minority and low-income populations surrounding these potential spent nuclear fuel management sites. Nevertheless, although the characterizations differ, the radiological impacts resulting from the proposed action under all alternatives present very low risk to the population as a whole. Therefore, no disproportionately high and adverse effects would be expected for any particular segment of the population, including minority and low-income populations, regardless of which set of definitions and assumptions were applied.

Implementation of the proposed action would have extremely low nonradiological effects on the environment at interim management sites, including the social and economic status of the general population, minority populations, and the low-income population surrounding interim management sites. Economic benefits that would result from increased cargo handling, transportation, and storage at interim management sites would be extremely small for the general population or any particular segment of the population residing near interim management sites.

4.2.4.6 Mitigation Measures at the Potential Foreign Research Reactor Spent Nuclear Fuel Management Sites

Based on the analyses of the environmental consequences for each potential foreign research reactor spent nuclear fuel management site included in Section F.4 of Appendix F, no mitigation measures would be necessary since all potential environmental impacts are substantially below acceptable levels or promulgated standards. However, each potential site would follow operation practices that would minimize the impacts in such areas as pollution prevention, cultural and ecological resources, ground and surface water quality, air quality, noise, traffic, operational and public health and safety, and accident prevention and mitigation. Descriptions of these practices are included in Appendix F, under Mitigation Measures for each site.

4.2.5 Short-Term Uses and Long-Term Productivity

Short-term impacts would be those associated with construction and operation of the storage facilities. No land would be used for the marine or ground transportation of foreign research reactor spent nuclear fuel. The use of land at the potential foreign research reactor spent nuclear fuel management sites would be in conformity with the land use policy of each site. The construction of new storage facility would lead to the loss of small acreage of terrestrial habitat. After adoption of an overall strategy for interim storage of all DOE-owned spent nuclear fuel (including spent fuel from foreign research reactors), some of the areas currently used for interim storage of spent nuclear fuel may be released for other productive uses

(DOE, 1995c). Ecological resources would be directly affected at the area of construction by land clearing. These resources would be limited to small mammals, reptiles, and songbirds. Given the small area that would be used, the overall effect would be of limited impacts on local populations and resources.

4.2.6 Irreversible and Irretrievable Commitments of Resources

The only irreversible use of resources during the marine and ground transportation of foreign research reactor spent nuclear fuel would be the use of petroleum fuel. Irreversible and irretrievable commitment to resources associated with management site activities are discussed below.

4.2.6.1 Management Site Resources

The irreversible and irretrievable commitment of resources resulting from the construction and operation of foreign research reactor spent nuclear fuel management site facilities would involve materials that could not be recovered or recycled, or resources that would be consumed or reduced to unrecoverable forms, including electrical energy, fuel, construction materials, and miscellaneous chemicals. Some construction materials are recyclable. Some of the resources would be irretrievable because of the nature of the commitment or the cost of reclamation. For example, human resources used for the construction and operation of the potential foreign research reactor spent nuclear fuel storage facilities would be irretrievably lost since these resources would be unavailable for use in other work activity areas. On the whole, foreign research reactor spent nuclear fuel management would not be particularly resource-intensive. The quantities of irreversible and irretrievable resources for each site are included in Appendix F, Section F.4.

4.2.6.2 Energy Resources

Under the basic implementation of Management Alternative 1, about 4.6 metric tons (5.1 tons) of highly-enriched uranium would be accepted into the United States. The energy content of this uranium would be equal to about 1.5 million megawatt-days or over 20 million barrels of No. 2 fuel oil if the conversion efficiency were 100 percent.

4.2.7 Impacts of Ultimate Disposition

Ultimate disposition of DOE's spent nuclear fuel, including foreign research reactor spent nuclear fuel, is a high priority. For planning purposes, DOE has determined that its spent nuclear fuel that is not otherwise managed (e.g., chemically separated, with the high-level waste being converted into a vitrified glass for repository disposal) is authorized for disposal in a geologic repository. Decisions regarding the actual disposition of DOE's spent nuclear fuel will follow appropriate review under the National Environmental Policy Act (NEPA).

It is possible that the foreign research reactor spent nuclear fuel could be accepted intact in a geologic repository. If DOE determines that geologic disposal of intact foreign research reactor spent nuclear fuel is possible, then there would be no onsite impacts beyond those associated with storage and packaging of the foreign research reactor spent nuclear fuel.

It is also possible that some form of processing could be necessary to convert the foreign research reactor spent nuclear fuel into a more stable form prior to its ultimate disposal. This processing could be a near-term new treatment technology, conventional chemical separation, or a new treatment technology that is implemented after an interim period of storage. DOE expects that any new treatment technology would produce no greater impacts than historical chemical separation activities. Therefore, the impacts of near-term treatment of the foreign research reactor spent nuclear fuel would be expected to be no greater than the impacts of chemically separating the same material as discussed in Section 4.3.6. If a new treatment technology is implemented after an interim period of storage and technology development, DOE expects that it would provide a substantial improvement over conventional chemical separation.

When disposal space is available, DOE would transport the intact or processed foreign research reactor spent nuclear fuel to a repository. This transportation would produce impacts similar to the ground transportation impacts discussed in Section 4.4.2.3. Handling and emplacement in the repository would produce impacts similar to those due to handling the spent nuclear fuel or processed waste at the DOE site because similar equipment and procedures would be used and the same regulatory limits on radiation doses would apply.

Yucca Mountain is the candidate site for a geologic repository for both spent nuclear fuel and high-level waste. Under the Nuclear Waste Policy Act, Congress found that a national problem had been created by the accumulation of spent nuclear fuel from commercial reactors and the accumulation of high-level waste. The Nuclear Waste Policy Act assigned to DOE the responsibility for managing the disposal of this spent nuclear fuel and high-level waste, specified the siting process, and authorized the construction of one geologic repository. Under the Nuclear Waste Policy Act Amendments Act of 1987, the process for selecting this repository was streamlined, and the Yucca Mountain site in Nevada was selected as the candidate site for a geologic repository.

Because the environmental documentation process for geologic disposal was established by the Nuclear Waste Policy Act, this EIS does not analyze environmental impacts of disposal at Yucca Mountain or alternative locations. After emplacement in a geologic repository, however, DOE expects there would be no more impacts to workers, the public, or the environment because the radioactive material would be effectively isolated.

In the event that a geologic repository were to be delayed, DOE assumed for purposes of this analysis that it would continue to manage the foreign research reactor spent nuclear fuel, or the high-level radioactive waste resulting from the chemical separation or other processing of such spent nuclear fuel, at the management sites until a geologic repository becomes available. The risk associated with this continued management is low and would not exceed the annual risk discussed in Section 4.2.4.1.

4.2.8 Summary of the Impacts of the Basic Implementation of Management Alternative 1

The principal impacts under the basic implementation of Management Alternative 1 would be occupational and public health and safety impacts. These are presented in Table 4-35 in terms of the risk of death due to cancer for each segment of the affected environment. It also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-35 shows that the greatest radiological risks would occur during ground transport or site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel produces a dose rate equal to the regulatory limit, (2) truck shipments expose people at highway rest stops for times about equal to the actual driving times, and (3) one individual at the DOE site receives the maximum dose allowed by DOE regulation (5,000 mrem) every year.

Table 4-35 Maximum Estimated Radiological Health Impacts of the Basic Implementation of Management Alternative 1

	Risks (LCF)	
	[illegible]	[illegible]

4.3 Implementation Alternatives of Management Alternative 1

As discussed in Chapter 2, a policy of managing foreign research reactor spent nuclear fuel in the United States could be implemented by various means. These variations on the basic implementation of Management Alternative 1 of the proposed action have been grouped into seven implementation alternatives. This section discusses their policy considerations and environmental impacts. For convenience, the seven implementation alternatives are listed briefly below:

1. Acceptance of amounts of material different from the amount in the basic implementation of Management Alternative 1,

2. Acceptance of foreign research reactor spent nuclear fuel for periods of time different from the period of time in the basic implementation of Management Alternative 1,

3. Implementation through financial arrangements different from those identified in the basic implementation of Management Alternative 1,

4. Taking title to the foreign research reactor spent nuclear fuel at locations different from the location in the basic implementation of Management Alternative 1,

5. Use of wet storage technology for the interim period instead of dry storage technology as in the basic implementation of Management Alternative 1,

6. Near term conventional chemical separation of the foreign research reactor spent nuclear fuel instead of interim storage as in the basic implementation of Management Alternative 1, and

7. Development and use of a new processing technology instead of interim storage as in the basic implementation of Management Alternative 1.

4.3.1 Implementation Alternative 1: Alternative Amounts of Spent Nuclear Fuel to be Accepted

DOE and the Department of State have evaluated the policy considerations and environmental impacts for different amounts of spent nuclear fuel and target materials under this implementation alternative.

4.3.1.1 Implementation Subalternative 1a: Accept Foreign Research Reactor Spent Nuclear Fuel Only From Developing Nations

Policy Considerations

Under this implementation subalternative, up to 1.9 MTHM and about 5,000 elements of foreign research reactor spent nuclear fuel would be accepted into the United States from developing nations (defined by the World Bank as nations with other-than-high-income economies). Up to about 238 kg (525 lb) of HEU would be removed from international commerce. By excluding developed countries, which generally share our nuclear weapons nonproliferation goals, but do not necessarily share our belief in the necessity for removing HEU from use in civil programs, this subalternative would have adverse consequences for U.S. nuclear weapons nonproliferation policy.

accept any near term shipments of spent nuclear fuel from developed countries, some reactor operators will be forced to either shut down their reactors or ship their spent nuclear fuel for reprocessing to the United Kingdom Atomic Energy Authority facility in Dounreay, United Kingdom, which is the only facility currently able and willing to reprocess foreign research reactor spent nuclear fuel. Operators in Belgium and Germany have already sent spent nuclear fuel elements to Dounreay for reprocessing. Since neither Dounreay nor any other facility is currently accepting aluminum-based research reactor spent nuclear fuel containing LEU for reprocessing, the only way a reactor operator can use reprocessing to control his spent nuclear fuel inventory is by using HEU for fuel. This could lead reactor operators to delay or cancel plans to convert to LEU, or, in some cases, to reconvert from LEU to HEU fuels.

The net result of reduced reliance on the United States is that foreign research reactor operators would be compelled to withdraw from the Reduced Enrichment for Research and Test Reactors (RERTR) program and continue operations on the HEU fuel cycle, with its lower costs and enhanced performance. Since the United States is barred from exporting HEU to virtually all foreign research reactors under the Energy Policy Act of 1992, operators would be forced to seek alternative suppliers of HEU, such as Russia and China. This could lead to renewed international commerce in weapons-usable HEU and undermine the U.S. nuclear weapons nonproliferation policy goal of seeking to minimize the civil use of HEU. Further, those countries that participated in the RERTR program considered U.S. acceptance of their spent nuclear fuel as a condition for incurring the substantial costs and technical difficulties of converting to LEU fuels. Failure to accept their spent nuclear fuel would jeopardize the nuclear weapons nonproliferation goals of the RERTR program and the reputation of the United States as a reliable partner in the conduct of international nuclear materials management.

There is another way this subalternative could undercut the RERTR program. The developing countries generally assess their technical capabilities by comparing themselves with the developed states of North America, Western Europe, and Japan. As noted above, one probable result of this subalternative is that more developed states will continue to use HEU-fueled research reactors, due to difficulty in reprocessing LEU spent nuclear fuel. If that happens, developing countries are likely to regard use of HEU-fueled reactors as more advanced and prestigious than LEU-fueled reactors, increasing the demand for such reactors as well as for HEU itself. Again, this would encourage increased stockpiles of HEU in various developed and developing countries, contrary to U.S. nuclear weapons nonproliferation policy.

If some countries are forced to shut down their reactors and thereby forego the medical and scientific benefits of these reactors, such a situation may lead to criticism that the United States is not a dependable nuclear partner. Some countries, including those in the developing world that have characterized the Treaty on the Non-Proliferation of Nuclear Weapons as a discriminatory bargain between the nuclear "haves" and the nonnuclear "have-nots," may be inclined to accuse the United States, fairly or unfairly, of having failed to comply with its Article IV Treaty pledge to facilitate "the fullest possible exchange of equipment, materials and scientific and technological information for the peaceful uses of nuclear energy." Actions that foster such negative perceptions would undoubtedly complicate the conferences which are scheduled to monitor compliance with the Non-Proliferation Treaty, and may complicate United States diplomatic efforts to attain other arms control and nuclear weapons nonproliferation objectives.

Marine Transport Impacts

Impacts of Incident-Free Marine Transport

The impacts of incident-free marine transportation were analyzed in the same manner as the basic implementation of Management Alternative 1. The incident-free transportation of spent nuclear fuel was estimated to result in total LCF that ranged from 0.008 to 0.009 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the ships' crews.

The range of impacts result from the analysis of shipment of the spent nuclear fuel on regularly scheduled commercial breakbulk vessels and on chartered container vessels and would be the same as for vessels analyzed in the evaluation of the basic implementation of Management Alternative 1. As in the basic implementation of Management Alternative 1, the difference between the two estimates is a result of the shorter vessel journey time for chartered vessels due to the intermediate port stops associated with the regularly scheduled commercial transport of the spent nuclear fuel.

Impacts of Accidents During Marine Transport

The consequences of the at-sea accidents for Implementation Subalternative 1a are no different than the consequences of at-sea accidents associated with the basic implementation of Management Alternative 1. For an accident involving the loss of a transportation cask in coastal waters, the maximum exposure to an individual is estimated to be 114 mrem per year. Due to the reduced number of cask shipments, the likelihood of such an accident would be reduced. Under this subalternative, 23 percent of the total number of cask shipments required under the basic implementation of Management Alternative 1 would be needed. The highest estimated risks due to an accident during marine transport would therefore be 0.00004 mrem per year peak dose to a human from the loss of a damaged cask in the deep ocean. This corresponds to an MEI risk of 1×10^{-10} LCF. This individual would have a chance of less than one in a billion of incurring an LCF due to an accident during marine transport.

Port Activity Impacts

Impacts of Incident-Free Port Activities

In the analysis of the basic implementation of Management Alternative 1, the radiological impact of port activities was estimated on a per-shipment basis. Implementation Subalternative 1a, accepting spent nuclear fuel from developing nations only, results in 23 percent of the total number of cask shipments that are required under the basic implementation of Management Alternative 1. The incident-free impacts of the port activities would be proportionally reduced. The estimated number of LCF associated with this subalternative range from 0.0008 to 0.003. As in the marine incident-free analysis, this range of impacts is the result of the analysis of two modes of spent nuclear fuel shipment, regularly scheduled commercial breakbulk vessels and chartered container vessels.

Impacts of Accidents During Port Activities

Port accident risks were calculated based on the per-shipment risks determined in the analysis of the basic implementation of Management Alternative 1. The analysis examined the impact of using a wide range of ports of entry based on the population around the port city, from high density population ports such as Elizabeth, NJ, to low-density ports such as the MOTSU terminal in North Carolina. The analysis also considered the impact of chartered shipments (no intermediate port stops before the vessel reaches the spent nuclear fuel port of entry) versus regularly scheduled commercial shipments with up to two intermediate ports of call before the spent nuclear fuel port of entry. The port accident risks over the entire program are estimated to range from 5 x 10^{-8} to 0.000004 LCF from radiation. The range of fatality estimates is due to both the differences in port city populations and the number of intermediate port stops.

The consequences of the maximum foreseeable port accident are identical to those of the basic implementation of Management Alternative 1. The frequency is lower due to the reduced number of shipments, so the MEI risk is reduced to 5 x 10^{-11} LCF.

Ground Transport Impacts

Impacts of Incident-Free Ground Transport

Radiological impacts of incident-free ground transportation were analyzed in the same manner for Implementation Subalternative 1a as for the basic implementation of Management Alternative 1. The results are presented in Figures 4-6 through 4-9. Incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.002 to 0.06 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The range of fatality estimates is caused by two factors: the option of using truck or rail to transport spent nuclear fuel and combinations of Phase 1 and Phase 2 potential foreign research reactor spent nuclear fuel management sites that created varying shipment numbers and distances.

The estimated number of radiation-related LCF for transportation workers ranged from 0.001 to 0.015. The estimated number of radiation-related LCF for the general population ranged from 0.0006 to 0.045, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.0002 to 0.01.

Impacts of Accidents During Ground Transport

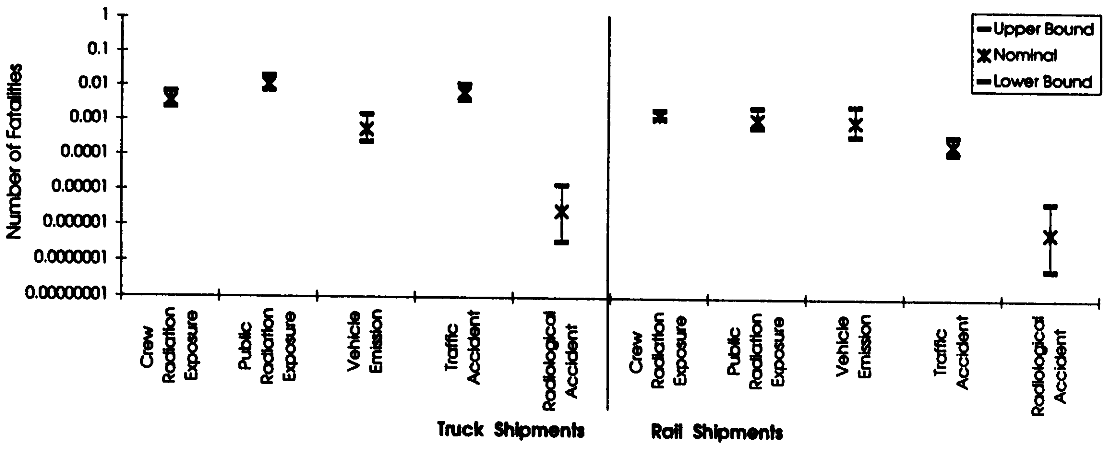
The transportation accident population risks over the entire program are estimated to range from 0.0000001 to 0.00006 LCF from radiation and from 0.0001 to 0.028 traffic fatalities, depending on the transportation mode and the potential foreign research reactor spent nuclear fuel management sites that might be selected. The reason for the range of fatality estimates is the same as those described for incident-free transportation.

The maximum foreseeable offsite transportation accident is identical to that for the basic implementation of Management Alternative 1. The risk is reduced to 2.7 x 10^{-12} LCF due to the reduced amount of ground transport.

Management Site Impacts

Impacts of Incident-Free Management Site Activities

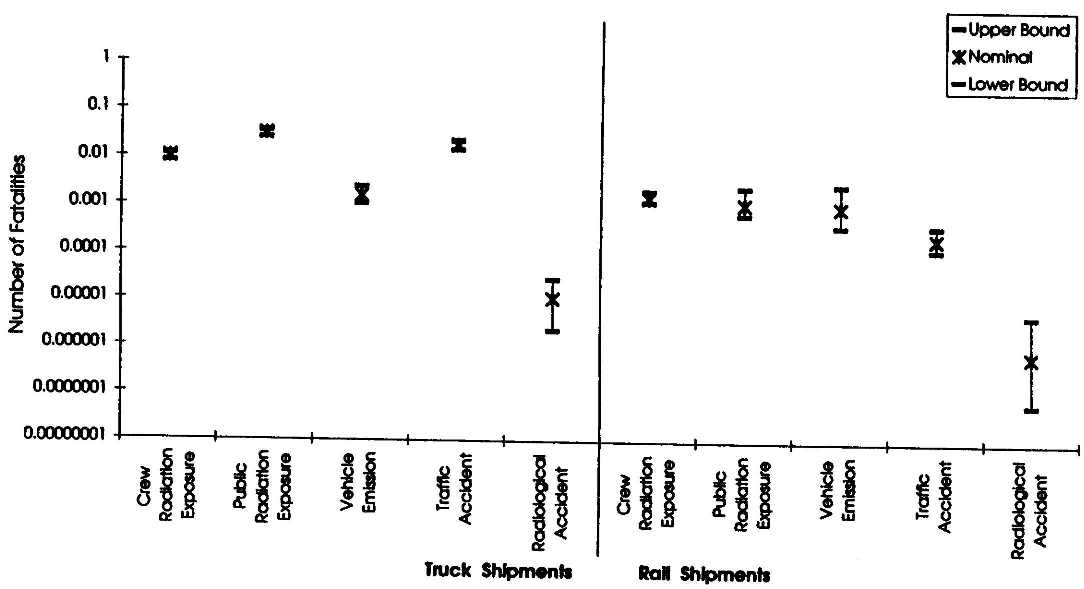
Impacts of incident-free site activities from Implementation Subalternative 1a are covered by the impacts from the basic implementation of Management Alternative 1. The maximally exposed worker radiation dose depends upon the duration of the receipts, not the amount of foreign research reactor spent nuclear fuel involved, and the duration in this subalternative is identical to the basic implementation of Management Alternative 1 (13 years). Thus, the maximally exposed worker dose is conservatively



- Same values apply to Programmatic SNF&INEL EIS 1992/1993 Planning Basis and INEL/SRS Regionalization alternatives

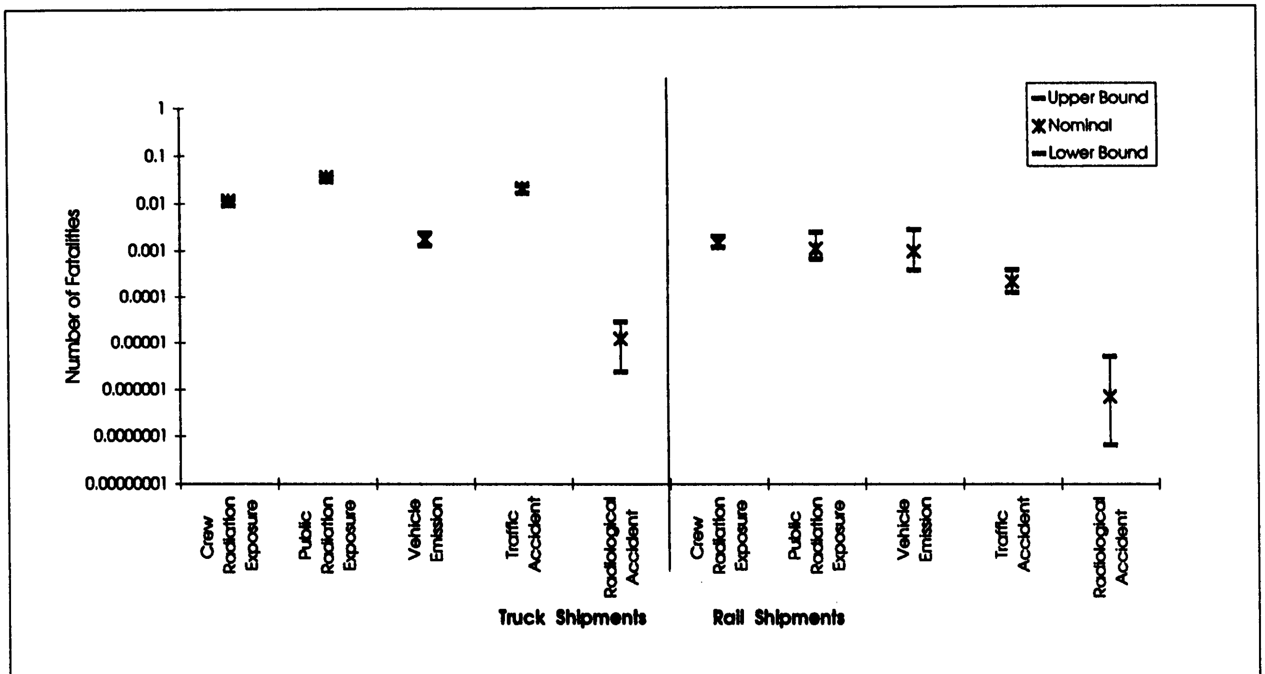
- Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities

Figure 4-6 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1a and the Programmatic SNF&INEL Final EIS Decentralization Alternative



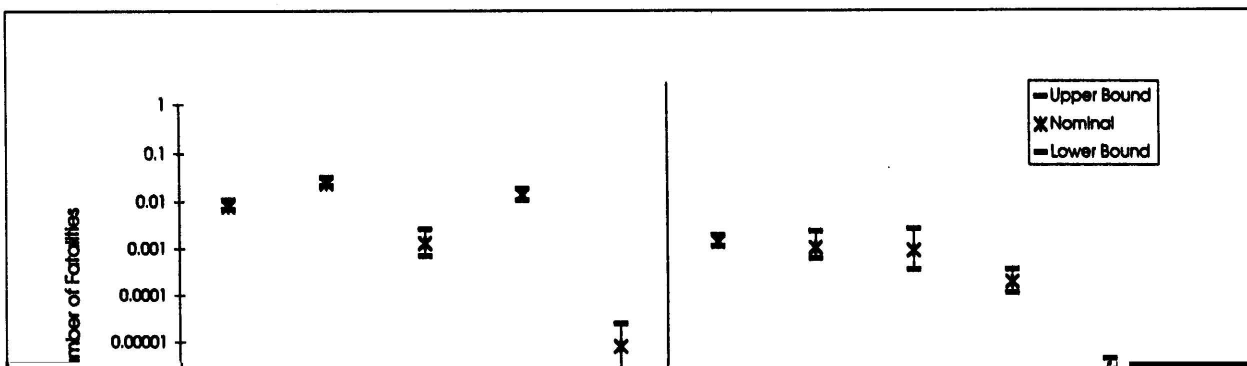
- Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities

Figure 4-7 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1a and the Programmatic SNF&INEL Final EIS Regionalization by Fuel Type Alternative



- Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities

Figure 4-8 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1a and the Programmatic SNF&INEL Final EIS Centralization to the Savannah River Site Alternative



assumed to be the same as in the basic implementation of Management Alternative 1. This would produce the maximally exposed worker risk identical to that in the basic implementation of Management Alternative 1 of 0.026 LCF.

The amount of foreign research reactor spent nuclear fuel that would be received and managed is 5,000 elements or approximately 22 percent of the number of elements in the basic implementation of Management Alternative 1. Thus, it is expected that the worker population risks at each management site would be approximately 22 percent of those calculated for the basic implementation of Management Alternative 1. The highest estimate of this risk under the basic implementation of Management Alternative 1 is 0.21 LCF, so the corresponding risk for this subalternative is 0.05, LCF, which is much less than one LCF.

Similarly, some of the incident-free public risk depends on the amount of foreign research reactor spent nuclear fuel involved and some depends on the duration of each activity. The risk that accrues during receipt and handling can be scaled down by the factor of 22 percent, while the risk that accrues during storage is dependent only on the duration of the storage. The highest estimated incident-free MEI risk in the basic implementation of Management Alternative 1 (1.4×10^{-7} LCF) is due to receipt and handling, so it is reduced by a factor of 22 percent to yield the corresponding risk of 3.1×10^{-8} LCF for this subalternative.

The highest estimated incident-free public population risk in Phase 1 of the basic implementation of Management Alternative 1 (0.00014 LCF) is due to storage, so it is not reduced in this subalternative. The corresponding Phase 2 risk (0.00013 LCF) is due to receipt and handling, so this component of the risk is reduced to 0.000029 LCF for this subalternative. The sum of the Phase 1 and Phase 2 risks is 0.00017 LCF.

Impacts of Accidents Onsite

The highest estimated MEI risk due to accidents in the basic implementation of Management Alternative 1 (0.0000034 LCF) is due to an accidental criticality in RBOF. This MEI risk is greater than any of the potential Phase 2 MEI risks, when those due to receipt/handling are reduced by the factor of 22 percent. Thus, the highest MEI risk due to accidents is 0.0000034 LCF.

The highest estimated population risk due to Phase 1 accidents in the basic implementation of Management Alternative 1 (0.096 LCF) is due to an accidental criticality in RBOF. The same facility could be used for the same period of time in this subalternative, so this component of the risk is unchanged. The corresponding Phase 2 risk (0.013 LCF) is due to receipt and handling, so it is reduced by the factor of 22 percent to 0.0029 LCF for this subalternative. The sum of the Phase 1 and Phase 2 risks is 0.099 LCF.

Summary of the Impacts of Implementation Subalternative 1a

The principal impacts under this subalternative would be occupational and public health and safety impacts. These are presented in Table 4-36 in terms of the risk of death due to cancer during each of the four segments of the affected environment. It also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. Each individual risk expresses the probability that one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-36 Maximum Estimated Radiological Health Impacts of Implementation Subalternative 1a (Developing Nations Only)

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	*Population*	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.00052	0	0.009
Accidents	1×10^{-10}	much less than 0.000004	---
Port Activities			
Incident-Free	0.00052	0	0.003
Accidents	5×10^{-11}	0.000004	---
Ground Transport			
Incident-Free	0.00052	0.045	0.015
Accidents	2.7×10^{-12}	0.00006	---
Site Activities			
Incident-Free	0.026	0.00017	0.05
Accidents	0.0000034	0.099	---
Highest Individual Risk			
Incident-Free	0.026	----	----
Accidents	0.0000034	----	---
Total Population Risk			
Incident-Free	----	0.045	0.077
Accidents	----	0.099	----

Table 4-36 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel produces a dose rate equal to the regulatory limit, (2) truck shipments exposes people at highway rest stops for times about equal to the actual driving times, and

4.3.1.2 Implementation Subalternative 1b: Accept Only Foreign Research Reactor Spent Nuclear Fuel that Contains HEU

Policy Considerations

Under this implementation subalternative, up to about 4.6 MTHM and 11,200 elements of foreign research reactor spent nuclear fuel would be accepted into the United States. All of this foreign research reactor spent nuclear fuel would contain HEU that was enriched in the United States.

Although this implementation subalternative would remove up to about 4.6 metric tons (5.1 tons) of HEU from international commerce, it almost certainly would result in the end of the RERTR program. As discussed in Chapter 1, the foreign research reactor operators have stated that they would not participate in the RERTR program unless the United States accepts their spent nuclear fuel, including LEU spent nuclear fuel. Otherwise, many research reactor operators would be likely to insist on using HEU fuel in their reactors in the future, which would increase international commerce in HEU. The most likely suppliers of this HEU would be Russia and China. DOE and the Department of State believe that in the long run, this subalternative would be contrary to the broader U.S. policy of nuclear weapons nonproliferation. Therefore, this subalternative is not analyzed in detail for environmental impacts in this EIS.

Summary of the Impacts of Implementation Subalternative 1b

Since the number of elements in this subalternative is about half the number of elements in the basic implementation of Management Alternative 1, the impacts would be roughly half of those calculated for the basic implementation of Management Alternative 1 (see Section 4.2.8).

4.3.1.3. Implementation Subalternative 1c: Accept Target Material in Addition to Foreign Research Reactor Spent Nuclear Fuel

Policy Considerations

This implementation subalternative would entail the shipment to the United States of not only HEU and LEU spent nuclear fuel, but of residual material from the production of molybdenum-99 for medical purposes. Molybdenum-99 is produced by the irradiation of targets in a research reactor. The targets are physically similar to the fuel for foreign research reactors. After being irradiated in a reactor, the targets are dissolved in acid to recover the molybdenum, leaving residual material containing enriched uranium. The United States has supplied HEU to Canada, Belgium, Argentina, and Indonesia for use as targets in the production of medical isotopes. The NRU reactor in Canada produces nearly all radioisotopes used in nuclear medicine in the United States.

This subalternative involves the acceptance of the following amounts of target material from these countries:

Canada	0.525 MTHM
Belgium	0.029 MTHM
Argentina	0.0011 MTHM
Indonesia	0.0014 MTHM
Total	0.5565 MTHM

This total has been rounded up to 0.6 MTHM for the purpose of analysis in this EIS. Under this subalternative, about 216 kg (476 lb) of HEU from target material would be removed from international commerce. This would be in addition to the estimated 4.6 metric tons (5.1 tons) of HEU that would be removed from international commerce under the basic implementation of Management Alternative 1.

Because the residual material contains weapons-usable HEU, there is a strong nuclear weapons nonproliferation rationale for including it in the scope of the management policy. This course of action would be desirable from a nuclear weapons nonproliferation standpoint, since it would leave the United States in control of the disposition of foreign research reactor spent nuclear fuel containing HEU, as well as residuals from the production of molybdenum-99, thereby minimizing the risk that such material might be diverted to a nuclear weapons program. This subalternative removes the most HEU from international civil commerce and provides the most support to U.S. nuclear weapons nonproliferation policy.

Furthermore, this subalternative would give the molybdenum-99 producers an incentive to switch from HEU targets to LEU targets. Appropriate LEU targets are currently under development as part of the RERTR program, and this target material would be accepted under this subalternative subject to the same conditions as the LEU foreign research reactor spent nuclear fuel in the basic implementation of Management Alternative 1.

The target material may be transported in one of two solid powder forms—as a calcine or an oxide. The calcine form would require about 140 cask shipments, while the oxide form would require about 57 cask shipments. The incident-free and accident risks are different for each form. The calcine material would produce an estimated 2.5 times more incident-free risk, but an estimated 10 times less accident risk than the oxide material. Furthermore, for transporting target material (unlike spent nuclear fuel), the accident risks would be greater than the incident-free risks. Therefore, to estimate conservative radiological risks, DOE and the Department of State assumed the target material would be transported as an oxide powder.

Marine Transport and Port Activities Impacts

The acceptance of target material would cause a very minor change in the marine and port incident-free impacts calculated for the basic implementation of Management Alternative 1. Up to only 7 cask shipments of oxide target material (6 from Belgium and 1[1] from Argentina or Indonesia), excluding the shipments from Canada, are estimated to be needed. This is less than one percent of the marine cask shipments of all foreign research reactor spent nuclear fuel in the basic implementation of Management Alternative 1. The incident-free impact per shipment is also reduced because the dose rate resulting from a cask loaded with the target material is expected to be much lower than that resulting from a cask loaded with foreign research reactor spent nuclear fuel.

For accident conditions, DOE and the Department of State estimated the risk due to an accident in an east coast port. The risk during marine transport would be much lower than the risk during port activities. The population risk due to accidents during port activities with seven casks of oxide target material is estimated to be 3.2×10^{-9} LCF. This is much lower than the population risk due to accidents with the foreign research reactor spent nuclear fuel.

The MEI risk is estimated to be 2.9×10^{-10} LCF, which is somewhat higher than the corresponding risk for the foreign research reactor spent nuclear fuel, but still very low.

1 Argentina or Indonesia would not produce enough target material to fill a transportation cask. In all likelihood, the target material from these countries would be shipped along with research reactor spent nuclear fuel elements.

Ground Transport Impacts

Impacts of Incident-Free Ground Transport

The impacts of incident-free ground transportation of target material were analyzed in the same manner as for the basic implementation of Management Alternative 1, except that, based on the low activity of the target material, the maximum dose rate at a distance of 2 m (6.6 ft) from the vehicle is estimated to be 0.1 mrem per hour. The risks calculated in this section could be added to those associated with foreign research reactor spent nuclear fuel transport. The incident-free transportation of target material was estimated to result in total latent fatalities that ranged from 0.0002 to 0.003 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew. When the risks of transporting target material are added to the risks of transporting the foreign research reactor spent nuclear fuel, the highest estimate of the population risk is 0.30 LCF.

The range of fatality estimates was due to two factors: the option of using truck or rail to transport target material and combinations of Phase 1 and Phase 2 sites that created varying shipment numbers and distances.

The estimated number of radiation-related LCF for transportation workers ranged from 0.00007 to 0.00074. The estimated number of radiation-related LCF for the general population ranged from 0.00015 to 0.0023, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.0001 to 0.004.

The impacts of transportation related to target material are summarized in Figures 4-10 through 4-13 and are described in more detail in Appendix E.

Impacts of Accidents During Ground Transport

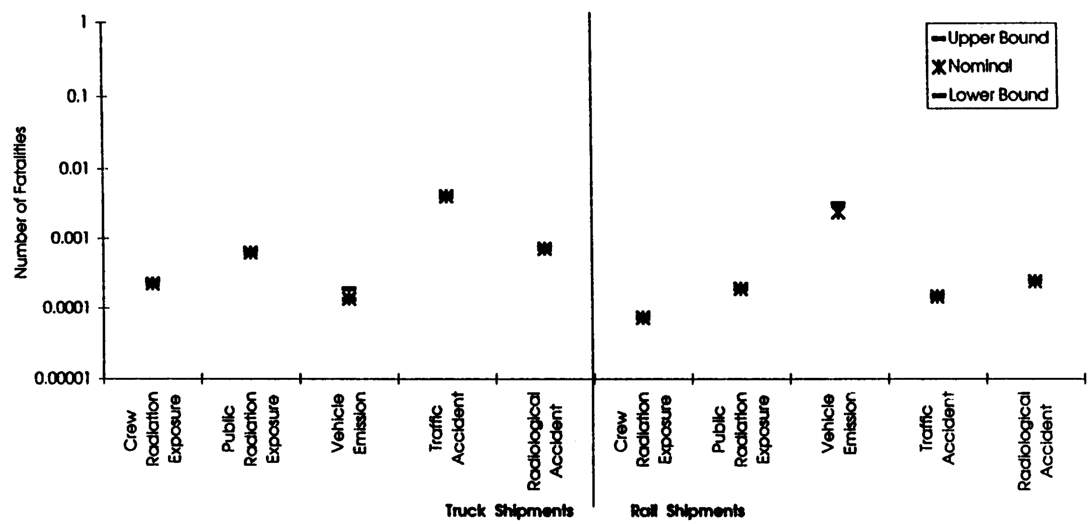
Cumulative transportation accident risks for the target material program are estimated to range from 0.0002 to 0.0054 LCF from radiation and from 0.0001 to 0.013 for traffic fatality, depending on the transportation mode and the potential foreign research reactor spent nuclear fuel management sites that might be selected. The reason for the range of fatality estimates is the same as those described for incident-free transportation. The highest estimate of the population risk due to accidents involving target material (0.0054 LCF) is higher than the same risk involving foreign research reactor spent nuclear fuel (0.00028 LCF). This difference is due to the physical/chemical forms of the two substances. Adding these two risks together yields the population risk due to accidents under Implementation Subalternative 1c, 0.0057 LCF.

The maximum foreseeable offsite transportation accident involves a cask shipment of powderized target material in a suburban population zone, and the risk is estimated to be 9.3×10^{-11} LCF to the MEI.

The impacts of transportation accidents are summarized in Figures 4-10 through 4-13, as described in the previous section, and are described in more detail in Appendix E. These tables can be used to assess the bounded absolute and relative risk of this subalternative under each representative Programmatic SNF&INEL Final EIS alternative.

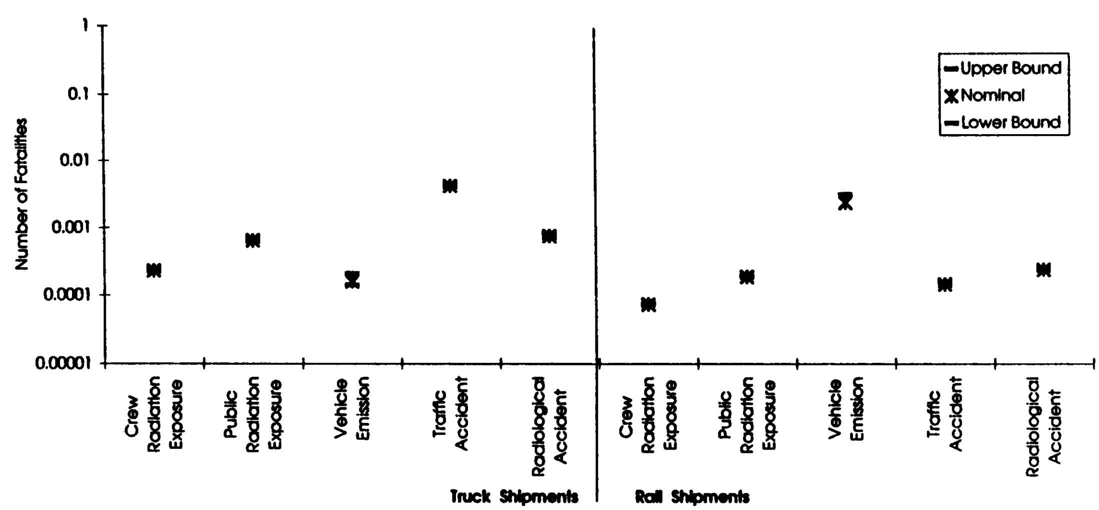
Management Site Impacts

There are two methods of preparing target material for transport. The first is calcining and canning the material with the aluminum included, and the second is to remove the aluminum from the solution, then oxidize and can the residue. Canned material from the first process has similar behavior as that of



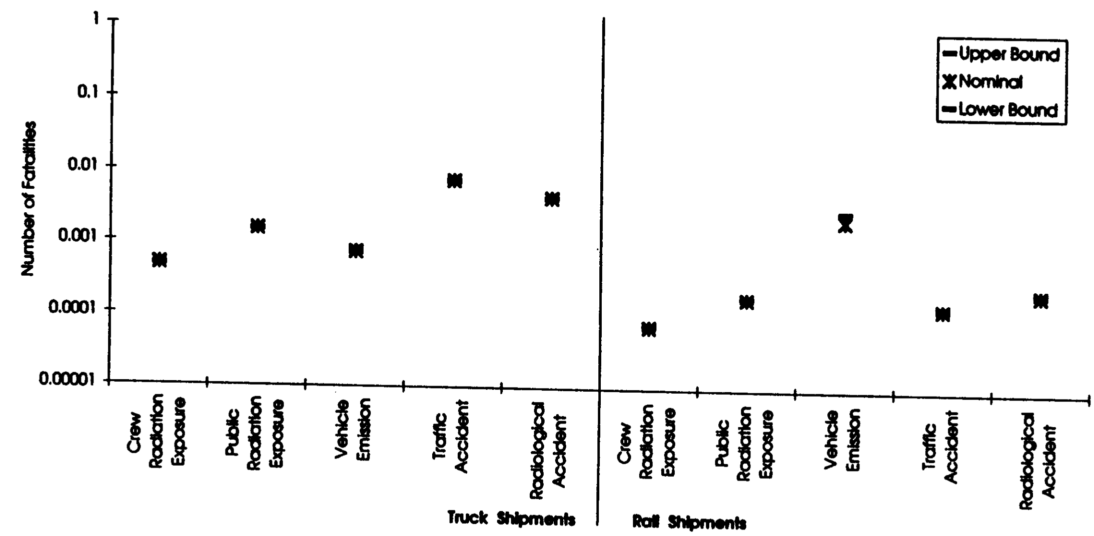
- *Same values apply to Programmatic SNF&INEL EIS 1992/1993 Planning Basis and INEL/SRS Regionalization alternatives*
- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-10 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1c and the Programmatic SNF&INEL Final EIS Decentralization Alternative



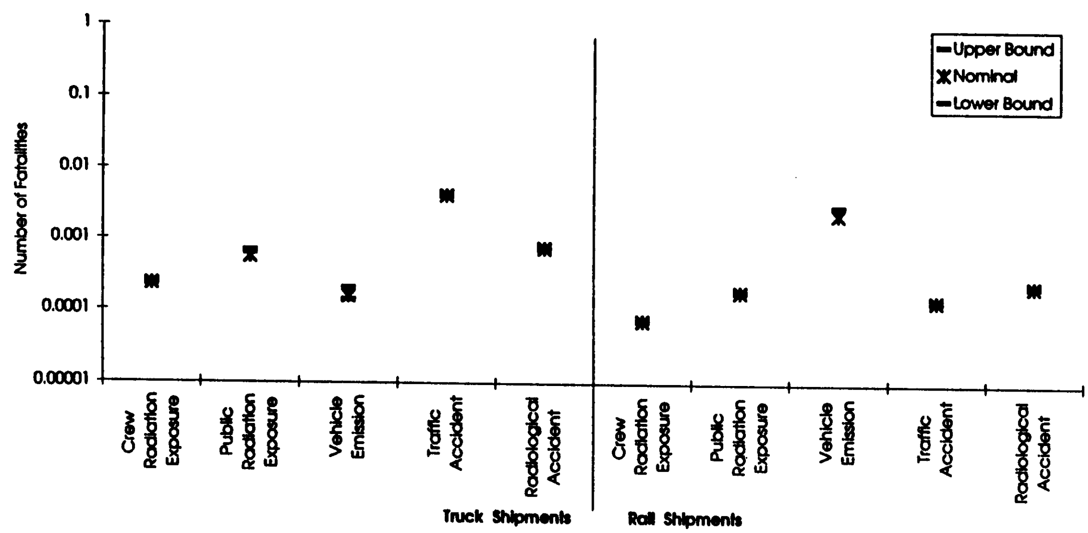
- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-11 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1c and the Programmatic SNF&INEL Final EIS Regionalization by Fuel Type Alternative



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-12 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1c and the Programmatic SNF&INEL Final EIS Centralization to the Savannah River Site Alternative



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-13 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1c and the Programmatic SNF&INEL Final EIS Centralization to the Idaho National Engineering Laboratory Alternative

aluminum-based foreign research reactor spent nuclear fuel containing about 40 g of uranium per can. The second process allows a higher amount of uranium, about 200 g, to be packed in the same size can. Use of the first process would result in 6,750 cans representing approximately 140 cask shipments. The second process would result in 1,350 cans representing approximately 57 cask shipments.

Target material cans would be stored like foreign research reactor spent nuclear fuel elements. The storage space required is a function of volume rather than the nuclear or thermal characteristics of the target material. On average, four cans of target material could be stored in the same space as one foreign research reactor spent nuclear fuel element. Therefore, the maximum storage required for target material (in the 40-gram cans) would be equivalent to 1,700 foreign research reactor spent nuclear fuel elements or approximately 7.4 percent of the space required for the foreign research reactor spent nuclear fuel elements under the basic implementation of Management Alternative 1. The storage facilities analyzed for the basic implementation of Management Alternative 1 include this margin in the sizing.

Impacts of Incident-Free Management Site Activities

Radioactive emissions would not be expected from the target material receipt or storage because this material contains no gaseous fission products. Therefore, the incident-free radiological impacts to the public would be the same as in the basic implementation of Management Alternative 1.

The collective dose to the crews that would handle the cask shipments would be 70 person-rem, assuming that the cans from 140 cask shipments would be placed in dry storage casks. The associated worker population risk would be 0.03 LCF. Adding this risk to the worker population risk of the basic implementation of Management Alternative 1 yields 0.24 LCF for the total incident-free worker population risk for Implementation Subalternative 1c.

Impacts of Accidents Onsite

The process by which target material is prepared for shipment (i.e., drying and canning of the solutions, see Appendix B, Section B.1.5) releases all gaseous fission products. In addition, the cans do not require any trimming when they arrive at a storage facility. A review of the hypothetical accident scenarios in the basic implementation of Management Alternative 1 indicates that only the aircraft crash with fire accident scenario would be applicable to target material. The cans are never cut, and there are no gaseous fission products, so the foreign research reactor spent nuclear fuel elements breach scenario would not be applicable. In addition, should an aircraft crash into the wet storage pool where the target material is stored, or if an accidental criticality in the pool were to occur, the radioactivity releases would be bounded by those of the spent nuclear fuel analyzed for these accidents. This is because the radioactive inventory per can is very small compared to that in the bounding foreign research reactor spent nuclear fuel.

A scenario involving an aircraft crash into a dry storage facility with an ensuing fire was analyzed for the target material. The scenario assumptions are similar to those described in Appendix F, Section F.6. Because of the size of each can, it was assumed that the transfer cask involved in the accident would contain 40 cans of target material containing maximum radionuclide inventories, i.e., that of 40 cans of 200 g of uranium per can cooled for at least 3 years.

The frequency of this event is estimated to be 3 percent of the 1×10^{-6} per year used in the accident analysis of the basic implementation of Management Alternative 1. This is because the number of transfer casks involving target material is less than 3 percent of that used for the approximately 22,700 elements in the basic implementation of Management Alternative 1. Therefore, the frequency of this scenario is less than 10^{-7} per year, and is considered to be non-foreseeable. Nonetheless, this accident was analyzed and its frequency is set conservatively at 10^{-7} per year. The analytical procedure was the same as that used in the basic implementation of Management Alternative 1.

The highest estimate of the MEI/NPAI accident risk with target material is 2.0 x 10^{-10} LCF, which would occur at the Oak Ridge Reservation (Table F-118, Appendix F). This risk is lower than the highest MEI/NPAI risk in the basic implementation of Management Alternative 1 (0.000010 LCF), so the risk for this subalternative is the same as in the basic implementation of Management Alternative 1. This hypothetical individual would still have one chance in one hundred thousand of incurring an LCF due to an accident on a site.

The highest estimate of the population risk with target material is 1.9 x 10^{-7} LCF, which also would occur at the Oak Ridge Reservation (Table F-118, Appendix F). To obtain the total population risk for this subalternative, this risk must be added to the corresponding risk from the basic implementation of Management Alternative 1 (0.11 LCF). The population risk due to accidents with target material is so small compared to the risk due to the foreign research reactor spent nuclear fuel that it makes essentially no contribution to the population risk for this subalternative. The population risk due to accidents under this subalternative would be the same as that under the basic implementation of Management Alternative 1.

Summary of the Impacts of Implementation Subalternative 1c

The principal impacts under this subalternative would be occupational and public health and safety impacts. These are presented in Table 4-37 in terms of the risk of death due to cancer during each of the four segments of this subalternative. It also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. The impacts of the basic implementation of Management Alternative 1 (Table 4-35) are added to the impacts of managing the target material to obtain the impacts of this subalternative. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-37 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE site receiving the maximum dose allowed by DOE regulation (5,000 mrem) every year.

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 1.4 x 10^{-7} LCF.

The highest estimated accident MEI risk is 0.000010 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than one in ten thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

Table 4-37 Maximum Estimated Radiological Health Impacts of Implementation Subalternative 1c (Target Material)

	Risks (LCF)
[illegible]	[illegible]

would accelerate the time at which the foreign research reactor operators and the governments of their host countries would become responsible for disposal of their own spent nuclear fuel. Up to about 4.1 metric tons (4.5 tons) of HEU would be removed from international commerce, which is about 0.5 metric tons (0.6 tons) less than under the basic implementation of Management Alternative 1.

This subalternative probably would not provide enough time for the foreign countries, especially the developing countries, to make arrangements for alternate means of managing their spent nuclear fuel. This could pressure various foreign research reactor operators to switch their reactors back to HEU fuel. In addition, it would probably, in effect, force many of the foreign research reactors with lifetime cores to shut down prematurely because it would be very difficult for them to find any means to dispose of their foreign research reactor spent nuclear fuel, other than to have DOE accept it.

Marine Transport Impacts

Impacts of Incident-Free Marine Transport

The impacts of incident-free marine transportation in the 5-year acceptance case were analyzed in the same manner as for the basic implementation of Management Alternative 1. The analysis was performed using the dose rates based on the exclusive-use regulatory limit for the shipment of spent nuclear fuel casks. The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.025 to 0.028 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the ships' crews.

The range of impacts results from the analysis of shipment of the spent nuclear fuel on regularly scheduled commercial breakbulk vessels and on chartered container vessels, and would be the same as for vessels analyzed in the basic implementation of Management Alternative 1. As in the basic implementation of Management Alternative 1, the difference between the two estimates is a result of the shorter vessel journey time for chartered vessels due to the intermediate port stops associated with the regularly scheduled commercial transport of the spent nuclear fuel.

The maximally exposed worker risk would be lower than that in the basic implementation of Management Alternative 1 due to the reduced acceptance period. The highest estimated maximally exposed worker risk would be 0.00032 LCF.

Impacts of Accidents During Marine Transport

The consequences of the at-sea accidents for Implementation Subalternative 2a are no different than the consequences of at-sea accidents associated with the basic implementation of Management Alternative 1. For an accident involving the loss of a transportation cask in coastal waters the maximum exposure to an individual is estimated to be 14,000 mrem per year. DOE and the Department of State would mitigate this impact, however, by recovering the cask. Due to the reduced number of cask shipments, the likelihood of such an accident would be reduced. Under this subalternative, approximately 81 percent of the total number of cask shipments required under the basic implementation of Management Alternative 1 would be needed. The highest risk to a human, expressed in terms of peak dose rate, would be 0.00015 mrem per year from the loss of a damaged cask in the deep ocean. Assuming an individual receives this dose for 5 years, the total MEI risk would be about 4×10^{-10} LCF.

Port Activity Impacts

Impacts of Incident-Free Port Activities

In the analysis of the basic implementation of Management Alternative 1, the radiological impact of port activities was estimated on a per-shipment basis. Implementation Subalternative 2a results in approximately 81 percent of the total number of cask shipments that are required in the basic implementation of Management Alternative 1. The incident-free impacts of the port activities would be proportionally reduced. The estimated number of LCF associated with this subalternative ranges from 0.0027 to 0.0098. As in the marine incident-free analysis, this range of impacts is the result of the analysis of two modes of spent nuclear fuel shipment, regularly scheduled commercial breakbulk vessels and chartered container vessels.

The maximally exposed worker risk would be lower than that in the basic implementation of Management Alternative 1 due to the reduced acceptance period. The estimated maximally exposed worker risk would be 0.00032 LCF.

Impacts of Accidents During Port Activities

Port accident risks were calculated based on the per-shipment risks determined in the analysis of the basic implementation of Management Alternative 1. The analysis examined the impact of using a wide range of ports based on the population around the port city, from high density population ports such as Elizabeth, NJ, to low-density ports such as the MOTSU terminal in North Carolina. The analysis also considered the impact of chartered shipments (no intermediate port stops before the vessel reaches the spent nuclear fuel port of entry) versus regularly scheduled commercial shipments with up to two intermediate ports of call before the spent nuclear fuel port of entry. The port accident risks over the entire program are estimated to range from 3×10^{-7} to 0.00002 LCF from radiation. The range of fatality estimates is due to both the differences in port city populations and the number of intermediate port stops.

The MEI risk would be lower than that of the basic implementation of Management Alternative 1 due to the reduced number of cask shipments. The highest estimated MEI risk is 1.6×10^{-10} LCF.

Ground Transport Impacts

Impacts of Incident-Free Ground Transport

The impacts of incident-free ground transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.010 to 0.27 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The range of fatality estimates was due to two factors: the option of using truck or rail to transport spent nuclear fuel and combinations of Phase 1 and Phase 2 management sites that created varying shipment numbers and distances.

The estimated number of radiation-related LCF for transportation workers ranged from 0.005 to 0.064. The estimated number of radiation-related LCF for the general population ranged from 0.005 to 0.20, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.001 to 0.041.

The maximally exposed worker risk would be lower than that in the basic implementation of Management Alternative 1 due to the reduced acceptance period. The highest estimated MEI risk would be 0.00032 LCF.

The impacts of transportation are summarized in Figures 4-14 through 4-17 and are described in more detail in Appendix E.

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire program are estimated to range from 0.000003 to 0.00026 LCF from radiation and from 0.001 to 0.13 for traffic fatality, depending on the transportation mode and the potential foreign research reactor spent nuclear fuel management sites that might be selected. The reason for the range of fatality estimates is the same as described for incident-free transportation.

The consequences of the maximum foreseeable offsite transportation accident are identical to those of the basic implementation of Management Alternative 1. The frequency is lower due to the reduced amount of ground transport, so the MEI risk is reduced to 1.1×10^{-11} LCF.

The impacts of transportation accidents are summarized in Figures 4-14 through 4-17, as described in the previous section, and are described in more detail in Appendix E. These tables can be used to assess the bounded absolute and relative risks of this subalternative under each representative Programmatic SNF&INEL Final EIS alternative.

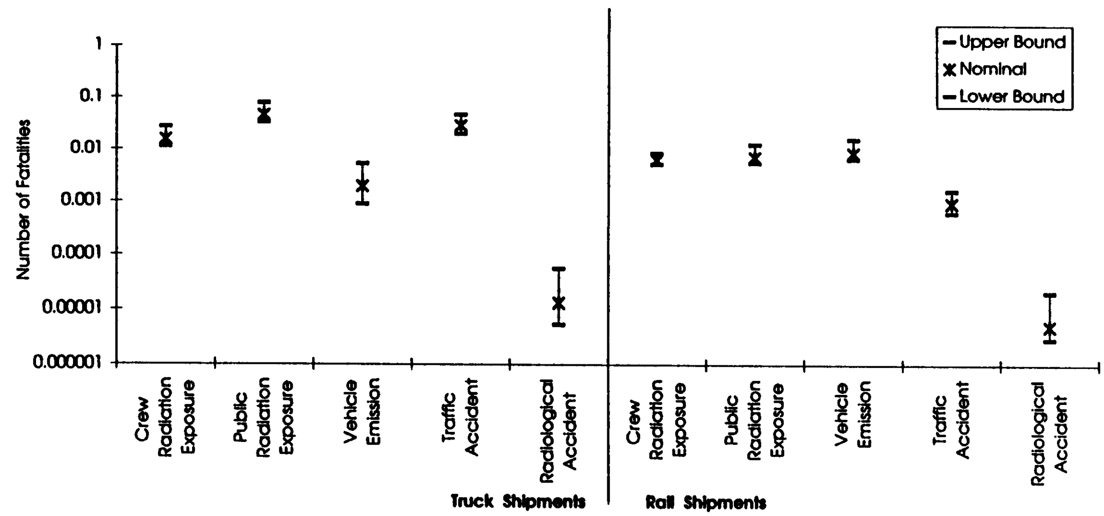
Management Site Impacts

As discussed in Chapter 2 of this EIS, Implementation Subalternative 2a reduces the quantity of foreign research reactor spent nuclear fuel to be managed to approximately 18,800 elements (compared to approximately 22,700 in the basic implementation of Management Alternative 1), but increases the rate of receipt to about 2,350 elements per year for an 8-year receipt period. This rate could challenge the capability of handling the incoming foreign research reactor spent nuclear fuel at a single site and could necessitate the use of both the Idaho National Engineering Laboratory and the Savannah River Site as near term foreign research reactor spent nuclear fuel management sites.

Incident-Free Impacts

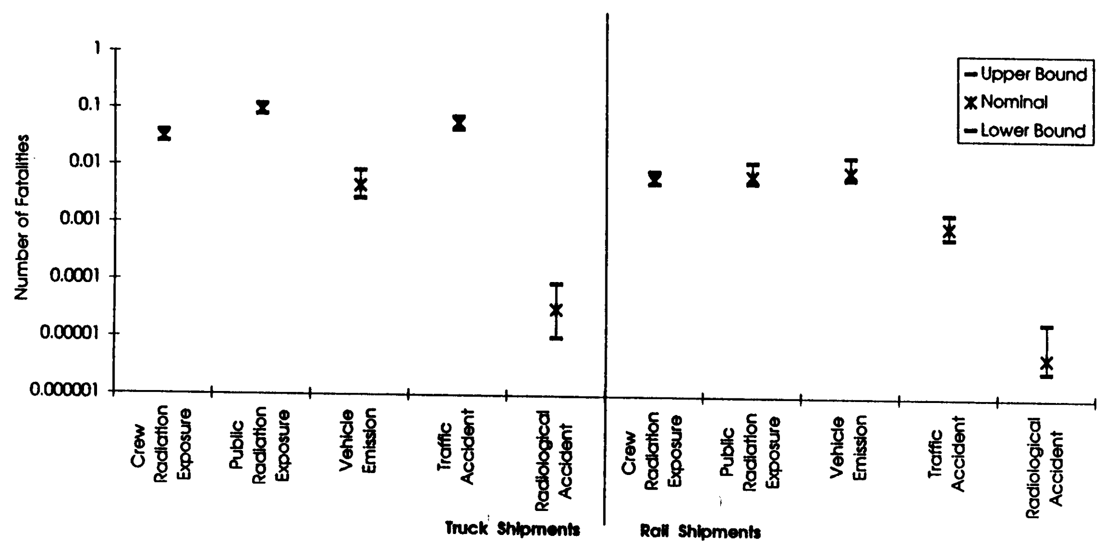
Based on the reduced number of foreign research reactor spent nuclear fuel elements that would be accepted under this subalternative, the worker population risk would be about 83 percent of that calculated for the basic implementation of Management Alternative 1. The maximally exposed worker risk was calculated in the same way as for the basic implementation of Management Alternative 1, with reduced handling time. If one worker received the maximum dose every year for eight years, his increased risk would be 0.016 LCF.

Some of the incident-free public risk depends on the amount of foreign research reactor spent nuclear fuel involved and some depends on the duration of each activity. The risk that accrues during receipt and handling can be scaled down by the factor of 83 percent from the basic implementation of Management Alternative 1, while the risk that accrues during storage is dependent only on the duration of the storage. The highest estimated incident-free public MEI risk in the basic implementation of Management Alternative 1 (1.4×10^{-7} LCF) is due to receipt and handling, so it is reduced by the factor of 83 percent to yield the corresponding risk for this subalternative (1.2×10^{-7} LCF).



- *Same values apply to Programmatic SNF&INEL EIS 1992/1993 Planning Basis and INEL/SRS Regionalization alternatives*
- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-14 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 2a and the Programmatic SNF&INEL Final EIS Decentralization Alternative



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-15 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 2a and the Programmatic SNF&INEL Final EIS Regionalization by Fuel Type Alternative

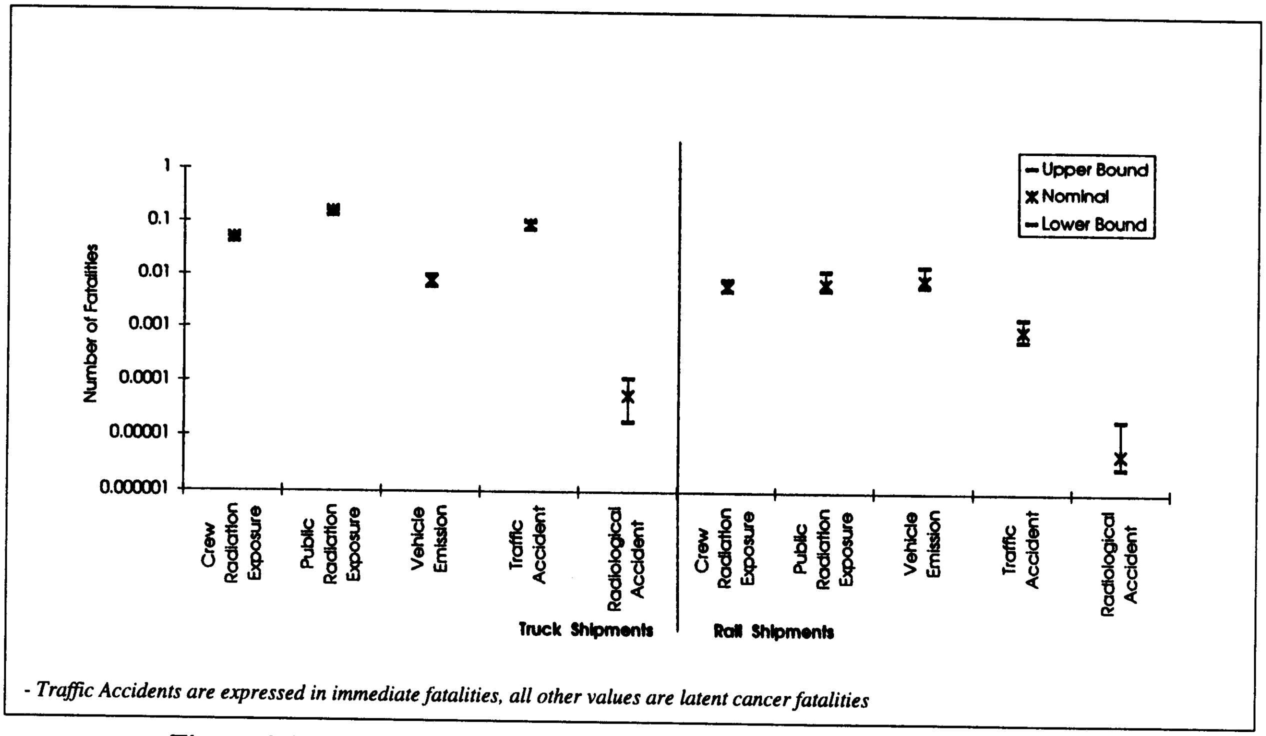


Figure 4-16 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 2a and the Programmatic SNF&INEL Final EIS Centralization to the Savannah River Site Alternative

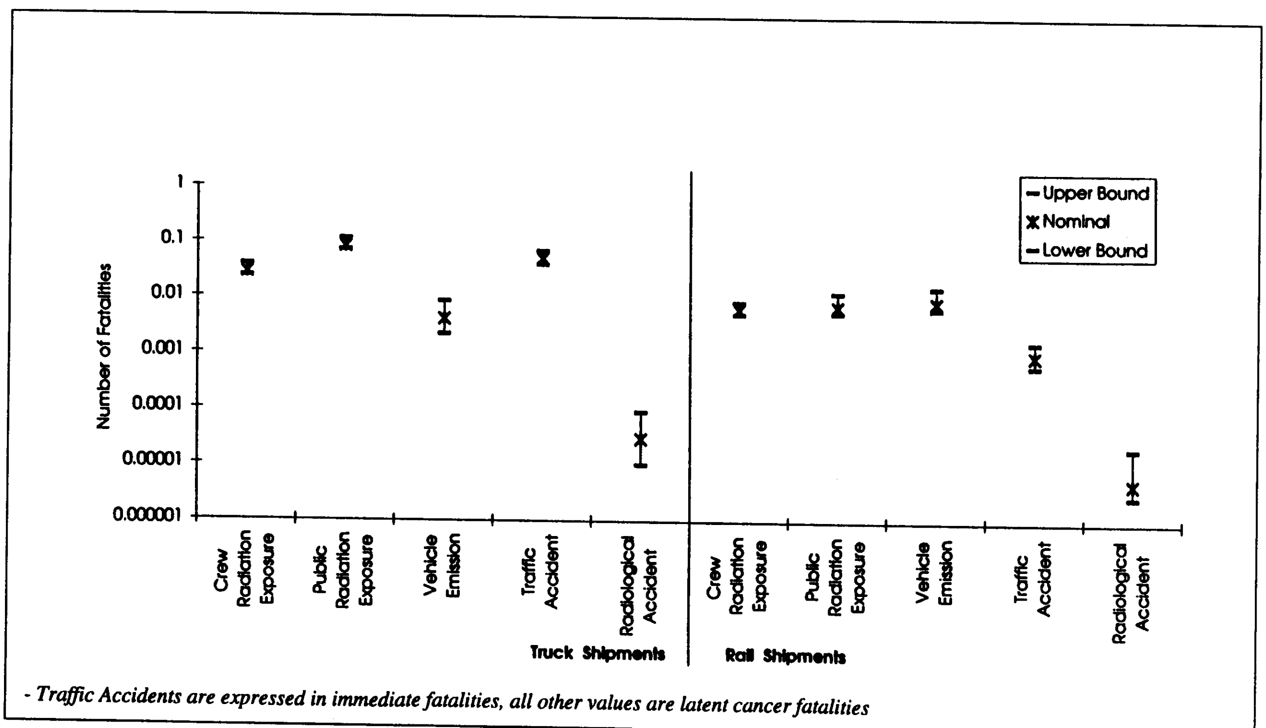


Figure 4-17 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 2a and the Programmatic SNF&INEL Final EIS Centralization to the Idaho National Engineering Laboratory Alternative

The highest estimated incident-free public population risk in Phase 1 of the basic implementation of Management Alternative 1 (0.00014 LCF) is due to 13 years of storage in L-Reactor Basin. The Phase 1 storage time in this subalternative would be slightly lower, and the estimated risk could be reduced, but for simplicity and to be conservative, DOE and the Department of State did not reduce this component of the risk estimate compared to the basic implementation. The corresponding Phase 2 risk (0.00013 LCF) is due to receipt and handling, so this component of the risk is reduced to 0.00011 LCF for this subalternative. The sum of the Phase 1 and Phase 2 risks is 0.00025 LCF.

Impacts of Accidents Onsite

The highest estimated public MEI risk due to accident conditions in the basic implementation of Management Alternative 1 (0.000010 LCF) is due to receipt and handling, so it is reduced by the factor of 83 percent to yield the corresponding risk for this subalternative (0.0000083 LCF). This is higher than any other combination of Phase 1 or Phase 2 annual risk and duration.

The highest estimated population risk due to Phase 1 accidents in the basic implementation of Management Alternative 1 (0.096 LCF) is due to an accidental criticality in RBOF. This facility would be used for less time in this subalternative and the estimated risk could be reduced, but for simplicity and to be conservative, DOE and the Department of State did not reduce this component of the risk estimate compared to the basic implementation. The corresponding Phase 2 risk (0.013 LCF) is due to receipt and handling, so this component of the risk is reduced by the factor of 83 percent, down to 0.011 LCF for this subalternative. The sum of the Phase 1 and Phase 2 risks is 0.11 LCF.

Summary of the Impacts of Implementation Subalternative 2a

The principal impacts under this subalternative would be occupational and public health and safety impacts. These are presented in Table 4-38 in terms of the risk of death due to cancer during each of the

Table 4-38 Maximum Estimated Radiological Health Impacts of Implementation Subalternative 2a (Five-Year Policy)

	Risks (LCF)	
	Maximum Exposed Worker	[illegible]

of the basic implementation of Management Alternative 1. Nevertheless, this subalternative would provide a mechanism whereby DOE and the Department of State could increase the amount of U.S. origin HEU that could be recovered.

Impacts

The environmental impacts would be the same as, or slightly less than, those of the basic implementation of Management Alternative 1. Delaying the acceptance of a small fraction of the total amount of foreign research reactor spent nuclear fuel accepted would have a miniscule effect on the results presented in Section 4.2.

4.3.3 Implementation Alternative 3: Alternative Financing Arrangements

Under the basic implementation of Management Alternative 1, DOE and the Department of State would subsidize developing nations and charge developed nations a competitive rate. As discussed in Chapter 2, DOE and the Department of State have identified three potential financial arrangements:

- Subsidize all nations,
- Charge all nations the full cost of managing their spent nuclear fuel, and
- Subsidize developing nations and charge developed nations the full cost of managing their spent nuclear fuel.

Policy Considerations

Subsidizing all countries would be the most expensive for the United States. All the costs of transport, handling, storage, preparation for disposal, and disposal would be borne by the United States. The amount of HEU that would be accepted under this arrangement would likely be the same as under the basic implementation of Management Alternative 1.

Charging all countries the full cost of foreign research reactor spent nuclear fuel management would be the least expensive for the United States. All the costs would be borne by the foreign countries. Many developing countries probably would be unable to pay these high costs and this could lead to large quantities of HEU foreign research reactor spent nuclear fuel remaining in the countries least able to protect it. This could also lead to charges, rightly or wrongly, that the United States was not complying with its obligations under Article IV of the Non-Proliferation Treaty. Even some developed countries might refuse to pay a full cost recovery fee, thus broadening the scope of problems this arrangement could cause.

Subsidizing developing countries and charging developed countries full cost of spent nuclear fuel management would be somewhat less expensive for the United States than the basic implementation of Management Alternative 1. Developing countries would be treated the same as in the basic implementation of Management Alternative 1, but developed countries would be charged more than in the basic implementation of Management Alternative 1. It is not clear how much more because the amount of a full cost recovery fee cannot be determined accurately at this time. Nevertheless, this increase over the internationally competitive rate could lead those nations which can reprocess to do so and perhaps to switch back to HEU fuel. Those nations in which reprocessing is not a viable option might force their reactors to shut down, and then charge, rightly or wrongly, that the United States was not complying with its obligations under Article IV of the Non-Proliferation Treaty.

Impacts

The different financial arrangements under this implementation alternative would have no direct effect on the environmental impacts of accepting and managing foreign research reactor spent nuclear fuel. Indirect effects are possible because, if the price is too high, some reactor operators may choose not to ship their spent nuclear fuel to the United States. This would reduce the amount of spent nuclear fuel accepted and thereby reduce the environmental impacts. It would be speculative, at best, to estimate the amount of spent nuclear fuel that might be excluded under this implementation alternative compared to the basic implementation of Management Alternative 1, so the changes in the environmental impacts cannot be quantified. It is clear however, that these changes would reduce overall environmental impacts in the United States during the policy period.

4.3.4 Implementation Alternative 4: Alternative Locations for Taking Title

Policy Considerations

The Price-Anderson Act applies to the shipments, independent of who holds title to the spent nuclear fuel. Thus, there is no change in the liability protection provided to the citizens of the United States, no matter where DOE takes title to the foreign research reactor spent nuclear fuel. Hence, there would be no change in the physical mode of shipping nor in the cost of shipping. Nevertheless, DOE and the Department of State are considering the following arrangements regarding the location for taking title to the foreign research reactor spent nuclear fuel:

- Taking title prior to shipment [i.e., at the foreign research reactor(s)],
- Taking title at the port(s) of entry, and
- Taking title at the DOE management site(s).

If DOE were to take title to the foreign research reactor spent nuclear fuel at the foreign research reactors, the liability protection afforded the citizens of the United States would not change, and the shipping arrangements would still be the same. However, DOE would then be liable for any mishaps that might occur in the foreign nations, or on the high seas. Thus, the potential liability to the United States might exceed the liability under the basic implementation of Management Alternative 1.

Taking title at the port(s) of entry would leave title in the hands of the foreign research reactor operators for the distance from the U.S. territorial waters limit to the port, thus potentially causing public concern about who would be liable to respond to any accident that might occur during that portion of the shipment. Similarly, taking title at the DOE management site would leave title in the hands of the foreign research reactor operators for an even greater distance within the United States, leading to even greater public concerns. These potential concerns would be borne of a misunderstanding because ownership does not affect shipping arrangements and precautions or liability protection. Nevertheless, it is likely that such concerns would exist.

Impacts

The environmental impacts (if any) of spent nuclear fuel shipments are not affected by the identity of the owner of the spent nuclear fuel. Therefore, the point of transfer of title is not a factor in determining environmental impacts.

4.3.5 Implementation Alternative 5: Wet Storage Technology for New Construction

Wet storage technology for new construction was considered instead of the dry storage technology contained in the basic implementation of Management Alternative 1, for all five potential foreign research reactor spent nuclear fuel management sites. The impacts during marine transport, port activities, and ground transport would be the same as in the basic implementation of Management Alternative 1. As in the basic implementation of Management Alternative 1, the analysis examined environmental topics including land use, socioeconomics, cultural resources, aesthetic and scenic resources, geology, air quality, water quality, ecology, occupational health and safety, noise, utilities and energy, and waste management.

The means by which this alternative would be implemented at each management site are presented in Sections 2.6.5.3.1 through 2.6.5.3.5. The environmental impact analysis assumes that a new wet storage facility, which is described in Section 2.6.5.1.2, would be constructed at the sites to receive and store foreign research reactor spent nuclear fuel after the Phase 1 period. At the Savannah River Site, the alternative could also be implemented at the Barnwell Nuclear Fuels Plant (BNFP) and at the Hanford Site by the addition of facilities to the WNP-4 Spray Pond. These facilities are described in Appendix F, Section F.3. The analysis parallels in all respects the impact analysis performed for the new dry storage facility of the basic implementation of Management Alternative 1. It is presented in detail in Appendix F, Section F.4, with methodology and assumptions for radiological impacts given in Sections F.5 and F.6.

As in the basic implementation of Management Alternative 1, the analysis showed that this implementation alternative would not cause any major environmental impacts. Further, none of the environmental topics would clearly differentiate among the potential foreign research reactor spent nuclear fuel management sites.

4.3.5.1 Occupational and Public Health and Safety

As in the basic implementation of Management Alternative 1 (see Section 4.2.4.1) radiological exposures are presented as emissions-related impacts, handling-related impacts, and accident-related impacts.

Impacts to the Public of Incident-Free Management Site Activities

Table 4-39 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at each Phase 2 site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

The highest estimated Phase 1 public MEI and population risks for this alternative are identical to those for the basic implementation of Management Alternative 1. All possible Phase 1 MEI risks are lower than the highest estimated Phase 2 MEI risk in the next paragraph, so they will drop out. The highest Phase 1 component of the population risk is 0.00014 LCF in the basic implementation.

Among all the potential Phase 2 foreign research reactor spent nuclear fuel management sites, the maximum annual dose to the public from emissions is 0.06 mrem per year and 0.06 person-rem per year at the Oak Ridge Reservation for the MEI dose and the population dose, respectively. If it is assumed that receipt of foreign research reactor spent nuclear fuel at the Oak Ridge Reservation could take place over a period of 3 years, the total MEI dose would be 0.18 mrem and the total population dose would be 0.18 person-rem. If it is further assumed that storage will continue for 30 years after the beginning of the receipt period, the total MEI dose from storage would be 1.4 x 10^{-5} mrem and the total population dose from storage would be 1.5 x 10^{-5} person-rem. The risks due to receipt and unloading would be much

Table 4-39 Annual Public Impacts for Receipt and Management of Foreign Research Reactor Spent Nuclear Fuel Under Implementation Alternative 5 (Wet Storage)

	MEI Dose (mrem/yr)	MEI Risk (LCF/yr)	Population Dose (person-rem/yr)	Population Risk (LCF/yr)
Savannah River Site				
Receipt/Unloading at:				
• BNFP	0.00065	3.3×10^{-10}	0.0045	0.0000023
• New Wet Storage Facility	0.00011	5.5×10^{-11}	0.0057	0.0000028
Storage at:				
• BNFP	7.5×10^{-9}	3.8×10^{-15}	4.8×10^{-8}	2.4×10^{-11}
• New Wet Storage Facility	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}
Idaho National Engineering Laboratory				
Receipt/Unloading at:				
• New Wet Storage Facility	0.00038	1.9×10^{-10}	0.0031	0.0000016
Storage at:				
• New Wet Storage Facility	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}
Hanford Site				
Receipt/Unloading at:				
• WNP-4 Spray Pond	0.00022	1.1×10^{-10}	0.0058	0.0000029
• New Wet Storage Facility	0.00020	1.0×10^{-10}	0.012	0.0000060
Storage at:				
• WNP-4 Spray Pond	5.9×10^{-10}	3.0×10^{-16}	1.6×10^{-8}	8.0×10^{-12}
• New Wet Storage Facility	8.8×10^{-10}	4.4×10^{-16}	6.9×10^{-8}	3.5×10^{-11}
Oak Ridge Reservation				
Receipt/Unloading at:				
• New Wet Storage Facility	0.060	3.0×10^{-8}	0.061	0.000031
Storage at:				
• New Wet Storage Facility	4.6×10^{-7}	2.3×10^{-13}	5.0×10^{-7}	2.5×10^{-10}
Nevada Test Site				
Receipt/Unloading at:				
• New Wet Storage Facility	0.00052	2.6×10^{-10}	0.00052	2.6×10^{-7}
Storage at:				
• New Wet Storage Facility	4.0×10^{-9}	2.0×10^{-15}	4.7×10^{-9}	2.4×10^{-12}

higher than those due to storage, so the maximum risk would be 0.18 mrem for the MEI and the sum of population doses would be 0.18 person-rem. The associated probabilities for incurring one LCF would be 9×10^{-8} LCF for the Phase 2 MEI risk and 0.00009 LCF for the Phase 2 population risk.

The maximum of the Phase 1 and Phase 2 incident-free public MEI risks is 9×10^{-8} LCF for this alternative. The sum of the Phase 1 and Phase 2 incident-free public population risks is 0.00023 LCF.

Impacts to Workers of Incident-Free Management Site Activities

As in the basic implementation of Management Alternative 1, workers would receive radiation doses during handling operations, such as receiving and unloading foreign research reactor spent nuclear fuel transportation casks at the site or transferring foreign research reactor spent nuclear fuel from one facility to another within the site. The methodology and assumptions for the analysis of this implementation alternative parallel that for the basic implementation of Management Alternative 1 as presented in Section 4.2.4.1 and Appendix F, Section F.5.

Table 4-40 presents the collective doses and risks that would be received by the members of the working crew, if that crew handled the total number of casks at the site.

Table 4-40 Handling-Related Impacts to Workers at Each Management Site Under Implementation Alternative 5 (Wet Storage)

Site	Worker Population Dose (person-rem)	Worker Population Risk (LCF)
Savannah River Site		
Phase 1: RBOF/L-Reactor Basin Phases 1 and 2: New Wet Storage Facility	250 360	0.10 0.14
Phase 1: RBOF/L-Reactor Basin Phases 1 and 2: BNFP	250 360	0.10 0.14
Phase 1: RBOF/L-Reactor Basin Phases 1 and 2: BNFP[a]	250 310	0.10 0.12
Idaho National Engineering Laboratory		
Phase 1: IFSF/CPP-749 Phases 1 and 2: New Wet Storage Facility	257 367	0.10 0.15
Phase 1: FAST Phases 1 and 2: New Wet Storage Facility	250 360	0.10 0.14
Hanford Site		
Phase 2: New Wet Storage Facility or WNP-4 Spray Pond	109	0.04
Oak Ridge Reservation		
Phase 2: New Wet Storage Facility	109	0.04
Nevada Test Site		
Phase 2: New Wet Storage Facility	109	0.04

[a] *Assumes that BNFP would be ready in 5 years instead of 10 years.*

As seen from Table 4-40, the maximum total collective dose to workers handling foreign research reactor spent nuclear fuel at a single site would be 367 person-rem for the case analyzed at the Idaho National Engineering Laboratory, which assumes that all foreign research reactor spent nuclear fuel is in dry storage during Phase 1 and is transferred to a new wet storage facility for Phase 2. The associated probability for one LCF among the working crew would be 0.15. The highest dose to working crews for both phases in more than one site is 366 person-rem: 109 person-rem at one of the three Phase 2 sites plus 257 person-rem at the Idaho National Engineering Laboratory as the Phase 1 site. The associated probability for developing one LCF among the working crews of the two sites is 0.15.

Accident-Related Impacts

The accident scenarios analyzed for this implementation alternative are the same as those analyzed for the basic implementation of Management Alternative 1.

Table 4-41 presents the frequency and consequences of the accidents analyzed for each management site for this implementation alternative. Multiplying the frequency of each accident times its consequences at each site and converting the radiation doses to LCF yields the annual risks associated with each potential accident at each candidate management site. Table 4-42 presents the annual risk estimates for wet storage.

The highest MEI or NPAI risk for Phase 1 would be the same as under the basic implementation of Management Alternative 1 (2.6×10^{-6} LCF). The highest annual MEI or NPAI risk for Phase 2 would be 0.000005 LCF per year, which is the annual risk to the NPAI from an accidental criticality at the Oak Ridge Reservation. Assuming that foreign research reactor spent nuclear fuel could be managed at the Oak Ridge Reservation for as long as 30 years, the Phase 2 component of this MEI/NPAI risk would

Table 4-41 Frequency and Consequences of Accidents at Each Management Site Under Implementation Alternative 5 (Wet Storage)

Site	Frequency (per yr)	Consequences							
		MEI		NPAI		Population		Worker	
		(mrem)	(LCF)	(mrem)	(LCF)	(person-rem)	(LCF)	(mrem)	(LCF)
Savannah River Site									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.0070	3.5×10^{-9}	0.00039	2×10^{-10}	0.23	0.00012	0.14	5.6×10^{-8}
• Accidental Criticality	0.0031	17	0.0000085	9.5	0.0000048	370	0.19	1,600	0.00064
• Aircraft Crash	1×10^{-6}	4.1	0.0000021	0.98	4.9×10^{-7}	150	0.075	400	0.00016
BNFP									
• Spent Nuclear Fuel Assembly Breach[a]	0.16	0.018	9×10^{-9}	0.00099	5×10^{-10}	0.028	0.000014	0.00080	3.2×10^{-10}
• Accidental Criticality[a]	0.0031	80	0.000040	75	0.000038	44	0.022	75	0.000030
• Aircraft Crash	1×10^{-6}	92	0.000046	31	0.000016	23	0.012	70	0.000028
Idaho National Engineering Laboratory									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.0016	8×10^{-10}	0.0036	1.8×10^{-9}	0.43	0.00022	0.14	5.6×10^{-8}
• Accidental Criticality	0.0031	28	0.000014	30	0.000015	140	0.070	1,800	0.00072
• Aircraft Crash	1×10^{-6}	22	0.000011	9.8	0.0000049	250	0.13	400	0.00016
Hanford Site									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.13	6.5×10^{-8}	0.0033	1.7×10^{-9}	1.6	0.00080	0.25	1.0×10^{-7}
• Accidental Criticality	0.0031	64	0.000032	14	0.000007	740	0.37	3,600	0.0014
• Aircraft Crash[b]	NA	NA	NA	NA	NA	NA	NA	NA	NA
WNP-4 Spray Pond									
• Spent Nuclear Fuel Assembly Breach[a]	0.16	0.15	7.5×10^{-8}	0.0033	1.7×10^{-9}	1.3	0.00065	0.00024	9.6×10^{-11}
• Accidental Criticality[a]	0.0031	97	0.000049	76	0.000038	620	0.31	120	0.000048
• Aircraft Crash[b]	NA	NA	NA	NA	NA	NA	NA	NA	NA
Oak Ridge Reservation									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.71	3.6×10^{-7}	0.20	1.0×10^{-7}	16	0.0080	0.68	2.7×10^{-7}
• Accidental Criticality	0.0031	1,500	0.00075	3,300	0.0017	1,400	0.70	6,800	0.0027
• Aircraft Crash	1×10^{-6}	380	0.00019	600	0.00030	2,900	1.5	1,900	0.00076
Nevada Test Site									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.054	2.7×10^{-8}	0.0016	8×10^{-10}	0.33	0.00017	0.10	4.0×10^{-8}
• Accidental Criticality	0.0031	88	0.000044	15	0.0000075	54	0.027	1,300	0.00052
• Aircraft Crash	1×10^{-6}	29	0.000015	4.2	0.0000021	61	0.031	290	0.00012

[a] *Emissions would be released through a tall stack, so workers would receive low doses.*

[b] *Aircraft crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

NA = Not applicable

Table 4-42 Annual Risks of Accidents at Each Management Site Under Implementation Alternative 5 (Wet Storage)

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
Savannah River Site				
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	5.5×10^{-10}	3.1×10^{-11}	0.000019	8.8×10^{-10}
• Accidental Criticality	2.7×10^{-7}	1.5×10^{-8}	0.00060	0.0000020
• Aircraft Crash	2.1×10^{-12}	4.9×10^{-13}	7.5×10^{-8}	1.6×10^{-10}
BNFP				
• Spent Nuclear Fuel Assembly Breach[a]	2.8×10^{-9}	8.0×10^{-11}	0.0000023	5.2×10^{-11}
• Accidental Criticality[a]	1.3×10^{-7}	1.2×10^{-7}	0.000070	9.2×10^{-8}
• Aircraft Crash	4.6×10^{-10}	1.6×10^{-11}	1.2×10^{-8}	2.8×10^{-10}
Idaho National Engineering Laboratory				
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	1.3×10^{-10}	2.9×10^{-10}	0.000035	8.8×10^{-9}
• Accidental Criticality	4.4×10^{-8}	4.7×10^{-8}	0.00022	0.0000022
• Aircraft Crash	1.1×10^{-11}	4.9×10^{-12}	1.3×10^{-7}	1.6×10^{-10}
Hanford Site				
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	1.1×10^{-8}	2.7×10^{-10}	0.00013	1.6×10^{-8}
• Accidental Criticality	1.0×10^{-7}	2.2×10^{-8}	0.0012	0.0000044
• Aircraft Crash[b]	NA	NA	NA	NA
WNP-4 Spray Pond				
• Spent Nuclear Fuel Assembly Breach[a]	1.2×10^{-8}	2.7×10^{-10}	0.00011	1.5×10^{-11}
• Accidental Criticality[a]	1.5×10^{-7}	1.2×10^{-7}	0.00096	1.5×10^{-7}
• Aircraft Crash[b]	NA	NA	NA	NA
Oak Ridge Reservation				
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	5.5×10^{-8}	1.6×10^{-8}	0.0013	4.4×10^{-8}
• Accidental Criticality	0.0000024	0.000005	0.0022	0.0000084
• Aircraft Crash	1.9×10^{-10}	3.0×10^{-10}	0.0000015	7.6×10^{-10}
Nevada Test Site				
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	4.2×10^{-9}	1.3×10^{-10}	0.000026	6.4×10^{-9}
• Accidental Criticality	1.4×10^{-7}	2.3×10^{-8}	0.000084	0.000016
• Aircraft Crash	1.5×10^{-11}	2.1×10^{-12}	3.1×10^{-8}	1.2×10^{-10}

[a] *Emissions would be released through a tall stack, so workers would receive low doses.*

[b] *Aircraft crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

NA = Not applicable

be 0.00015 LCF. This is higher than any other combination of Phase 2 annual accident risks and associated durations in this implementation alternative. Taking the maximum of the Phase 1 and Phase 2 MEI risks yields 0.00015 LCF for the maximum MEI risk due to accidents.

The highest population risk for Phase 1 would be the same as under the basic implementation of Management Alternative 1, 0.096 LCF. The highest annual population risk for Phase 2 would be 0.0022 LCF per year, which is the annual risk to the public from an accidental criticality at the Oak Ridge Reservation. Assuming that foreign research reactor spent nuclear fuel could be managed at the

Oak Ridge Reservation for as long as 30 years, the Phase 2 component of this population risk would be 0.066 LCF. This is higher than any other combination of Phase 2 annual accident risks and associated durations in this implementation alternative. Adding the Phase 1 and Phase 2 population risks yields 0.16 LCF for the total population risk due to accidents.

4.3.5.2 Topics Not Discussed in Detail

Nonradiological impacts associated with the wet storage implementation alternative are similar to those for dry storage considered in the basic implementation of Management Alternative 1. They are discussed in detail in Appendix F, Section F.4.

Impacts at each management site typically associated with construction activities such as land use, socioeconomics, cultural resources, aesthetic and scenic resources, air quality, ecology, and noise are similar because: (1) both dry and wet storage facilities could be constructed at the same locations at each site; and (2) both facilities are approximately the same size. As indicated in Section 2.6.5.1, the construction of the wet storage facility would disturb approximately 2.8 ha (7 acres) of land while the construction of the dry storage facility would disturb 3.6 to 4.5 ha (9 to 11 acres). Specifically for the Savannah River Site, if the wet storage alternative is implemented using the BNFP facility there would be no impacts associated with construction activities.

Impacts at each management site typically associated with the operation of the facilities such as air quality, water quality, socioeconomics, utilities, and waste generation are also very similar as indicated in Section 2.6.5.1. The only notable difference is indicated in water use. The wet storage facility would use 1.5 million liters (409,000 gal) per year during the storage mode of the operation (over 30 years) compared to 0.9 million liters (238,000 gal) per year used by the dry storage facility over the same period. This difference, however, is small compared to typical water consumption rates at the sites: 1.14 billion liters (300 million gal) per year at the Nevada Test Site to 88 billion liters (23.2 billion gal) per year at the Savannah River Site.

4.3.5.3 Summary of the Impacts of Implementation Alternative 5

The principal impacts under this implementation alternative would be occupational and public health and safety impacts. These are presented in Table 4-43 in terms of the risk of death due to cancer during each of the four segments of the affected environment. It also shows, in the bottom rows, the highest of the individual risks and the total population risks. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-43 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE site receiving the maximum dose allowed by DOE regulation (5,000 mrem) every year.

Table 4-43 Maximum Estimated Radiological Health Impacts of Implementation Alternative 5 (Wet Storage)

	Risks (LCF)	
	Maximally Exposed Worker,	Population

research reactor spent nuclear fuel. After some upgrading, the facilities at the Idaho National Engineering Laboratory would have the capability to chemically separate all the foreign research reactor spent nuclear fuel.

4.3.6.1 Implications of Chemical Separation for U.S. Nonproliferation Policy

As a matter of policy, the United States does not currently engage in reprocessing or chemical separation to extract plutonium for civilian or military purposes. U.S. policy is also not to encourage the civilian use of plutonium and to explore means to limit the stockpiling of plutonium from civil nuclear programs. This alternative nonetheless considers scenarios whereby the United States might engage in future chemical separation of foreign research reactor spent nuclear fuel. If a decision were made pursuant to this EIS to chemically separate some or all of the foreign research reactor spent nuclear fuel, the limited amount of plutonium in the spent fuel would not be separated. Rather it would be left in, and disposed of with, the high-level radioactive wastes produced during the chemical separation operation.

Two alternatives are evaluated for handling the highly enriched uranium in the spent fuel, either to blend it down to low enriched uranium (the preferred alternative, if any chemical separation is undertaken), or to separate it as HEU and place it in safe, secure storage. Chemical separation of foreign research reactor spent nuclear fuel, with blending down of the separated uranium, would, in fact, result in a reduction in the amount of HEU – a major goal of the U.S. Nuclear Weapons Nonproliferation Policy announced in September 1993. Despite this fact, there is a concern that other states may perceive only that the U.S. has restarted reprocessing.

For example, the potential exists that other states (e.g., Iran), might use the restart of reprocessing in the United States as an excuse to continue current programs or begin new ones – activities that would run counter to U.S. nuclear weapons nonproliferation interests. The implications in North Korea, where the United States has been actively working to create a nonreprocessing zone, as well as in other states, could complicate current U.S. nonproliferation activities.

4.3.6.2 General Assumptions and Analytic Approach

Potential impacts at the Savannah River Site and the Idaho National Engineering Laboratory were estimated separately. The impacts due to chemical separation and associated onsite activities would be in addition to those due to marine transport, port activities, and ground transport.

As discussed in Section 2.2.2.6, DOE and the Department of State have analyzed four possible chemical separation subalternatives under this implementation alternative. These four subalternatives, with spent nuclear fuel amounts and estimated facility run durations are:

	Amount (MTHM)	*Duration (Years)*
Savannah River Site (only aluminum-based spent nuclear fuel)		
• Foreign research reactor spent nuclear fuel only	18.2	13
• Foreign research reactor spent nuclear fuel plus other spent nuclear fuel	51	13
Idaho National Engineering Laboratory (aluminum-based and TRIGA spent nuclear fuel)		
• Foreign research reactor spent nuclear fuel only	19.2	12
• Foreign research reactor spent nuclear fuel plus other spent nuclear fuel	65	12

The duration of chemical separation operations dedicated to foreign research reactor spent nuclear fuel is driven by the rate of foreign research reactor spent nuclear fuel receipt at the Savannah River Site or the Idaho National Engineering Laboratory. The facility run durations at Savannah River Site are both up to 13 years, whether the facilities would be chemically separating only the 18.2 MTHM of foreign research reactor spent nuclear fuel or the 51 MTHM of spent nuclear fuel. Because the additional spent nuclear fuel would be chemically separated at the same time as the foreign research reactor spent nuclear fuel in a parallel process, only the combined impacts will be used to determine the risks associated with the overall operations. There are other nuclear materials, such as the Mark-31 targets currently stored at the Savannah River Site, which could also be chemically separated. These nuclear materials are not included in this implementation alternative, but they are covered under cumulative impacts. The impacts of running the facilities are based on conservative assumptions regarding incident-free annual emissions and possible accident releases which cover this range of throughputs.

The facility run durations at the Idaho National Engineering Laboratory are estimated to be up to 12 years. Furthermore, the same type of conservative assumptions regarding incident-free emissions and accidental releases are applied to calculate the environmental impacts.

As discussed in Section 2.2.2.6, the implementation component of uranium disposition has policy implications. The separated LEU could be returned to the commercial sector for reuse as reactor fuel. The HEU could be blended down to LEU or it could be processed directly to an oxide and stored. If a decision is made to chemically separate this spent nuclear fuel, it would be DOE's preference to blend down the HEU to LEU and thus preclude the possibility of this material ever being used in a nuclear weapon.

4.3.6.3 Marine Transport Impacts

The marine transport impacts of this implementation alternative would be identical to those of the basic implementation of Management Alternative 1, as discussed in Section 4.2.1.

4.3.6.4 Port Activities Impacts

The port activities impacts of this implementation alternative would be identical to those of the basic implementation of Management Alternative 1, as discussed in Section 4.2.2.

4.3.6.5 Ground Transport Impacts

The impacts due to ground transport of foreign research reactor spent nuclear fuel in this implementation alternative would be slightly lower than those of the basic implementation of Management Alternative 1, because the Phase 2 intersite shipments would not occur.

If the aluminum-based foreign research reactor spent nuclear fuel were chemically separated at the Savannah River Site it could not then be transported to another management site as spent nuclear fuel. The high-level waste resulting from this chemical separation would be managed onsite for the duration of the 40-year program period. The TRIGA foreign research reactor spent nuclear fuel would be transported to either of the two sites for management and it would not be transported again for the duration of the 40-year program period.

Similarly, if all the foreign research reactor spent nuclear fuel were chemically separated at the Idaho National Engineering Laboratory, it could not then be transported to another management site as spent nuclear fuel. The high-level waste resulting from this chemical separation would be managed onsite for the duration of the 40-year program period.

Impacts of Incident-Free Ground Transport

The impacts of incident-free ground transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.020 to 0.27 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The range of fatality estimates was due to two factors: the option of using truck or rail to transport spent nuclear fuel and combinations of management sites that created varying cask shipment numbers and distances.

The estimated number of radiation-related LCF for transportation workers ranged from 0.009 to 0.065. The estimated number of radiation-related LCF for the general population ranged from 0.011 to 0.21, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.003 to 0.05.

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire program are estimated to range from 0.000004 to 0.00014 LCF from radiation and from 0.002 to 0.13 for traffic fatality, depending on the transportation mode and the potential foreign research reactor spent nuclear fuel management sites that might be selected. The reason for the range of fatality estimates is the same as those described for incident-free transportation.

The consequences of the maximum foreseeable offsite transportation accident are identical to those of the basic implementation of Management Alternative 1. The frequency is lower due to the reduced amount of ground transport, so the MEI risk is reduced to 1.3×10^{-11} LCF.

4.3.6.6 Impacts at the Potential Foreign Research Reactor Spent Nuclear Fuel Management Sites

DOE and the Department of State evaluated near term chemical separation at the Savannah River Site and the Idaho National Engineering Laboratory for five key types of impacts: (1) Socioeconomics, (2) Air Quality, (3) Water Quality, (4) Occupational and Public Health and Safety, and (5) Waste Management. The other impacts are all the same as those described in the basic implementation of Management Alternative 1. The analytic approach was to use the results published in the Programmatic SNF&INEL Final EIS (DOE, 1995c) and the Interim Management of Nuclear Materials Final EIS (DOE, 1995a) whenever possible. Usually, these results can be adopted directly.

4.3.6.6.1 Socioeconomics

Savannah River Site

The chemical separation facilities at the Savannah River Site were last operated in 1992. The facilities are in a warm standby condition and are currently fully staffed. Use of these facilities would not have a notable net impact upon employment or the regional economy.

Idaho National Engineering Laboratory

The chemical separation facilities at the Idaho National Engineering Laboratory were last operated in 1990 and are currently in the process of being cleaned out in preparation for decommissioning. Some staff would need to be added eventually, but the use of these facilities would not have a notable net impact upon employment or the regional economy.

4.3.6.6.2 Air Quality

Savannah River Site

Incident-Free Nonradiological Emissions

DOE has analyzed the expected nonradiological emissions from its chemical separations facilities at the Savannah River Site in the Programmatic SNF&INEL Final EIS (DOE, 1995c). All estimated emissions would be small increases over baseline site-wide totals and within regulatory limits (DOE, 1995c).

Incident-Free Radiological Emissions

DOE has analyzed the expected airborne radiological emissions from the Savannah River Site chemical separations facilities in the Interim Management of Nuclear Materials EIS (DOE, 1995a). These radiological emissions are presented in Table 4-44 (Grainger, 1995). The health effects from these airborne emissions are discussed in Section 4.3.6.6.4 below.

Table 4-44 Annual Incident-Free Airborne Radiological Emissions at the Savannah River Site that Contribute to the Offsite Dose[a]

Element	Ci/yr
Tritium	57.8
Cesium-134	0.002
Cesium-137	0.12
Curium-242/244	0.12
Cerium-144	0.0059
Americium-241	0.016
Cobalt-60	0.000000053
Plutonium-238	0.078
Plutonium-239	0.020
Strontium-89/90	0.17
Iodine-131	0.0053
Uranium-235/238	0.039
Antimony-125	0.018
Ruthenium-106	0.20

[a] *Krypton-85 would be released at an estimated rate of 120,000 Ci/yr*

Source: Grainger, 1995

Krypton-85 emissions are not included in Table 4-44 because these releases are not normally measured or calculated. The health effects resulting from krypton-85 releases are very low compared to those resulting from other isotopes that are being measured. Krypton is an inert gas with no affinity for biological systems, so it does not adhere to the lungs if inhaled. The radioactive isotope of krypton would cause such a low level of harm to the population near the Savannah River Site because it remains in the human body for only very brief periods of time. The total amount of krypton-85 that would be contained in all of the aluminum-based foreign research reactor spent nuclear fuel is conservatively estimated to be 1.5 x 10^6 curies. Assuming this is released gradually over the 12-year reprocessing period, the annual emission rate would be 1.2 x 10^5 curies per year.

Idaho National Engineering Laboratory

Incident-Free Nonradiological Emissions

DOE has analyzed the expected nonradiological emissions from its chemical separations facilities at the Idaho National Engineering Laboratory in the Programmatic SNF&INEL Final EIS (DOE, 1995c). All estimated emissions are within regulatory limits (DOE, 1995c).

Incident-Free Radiological Emissions

DOE has also analyzed the expected radiological emissions from the Idaho National Engineering Laboratory chemical separations facilities in the Programmatic SNF&INEL Final EIS (DOE, 1995c). These are presented in Table 4-45. The radiological emission rates were estimated using conservative engineering calculations based on knowledge of the proposed activity. These emission rates are representative of emissions that could occur during Implementation Alternative 6 at the Idaho National Engineering Laboratory. Human health consequences are discussed in Section 4.3.6.6.4.

Table 4-45 Annual Incident-Free Airborne Radiological Emissions at the Idaho National Engineering Laboratory

Element	Ci/yr
Tritium + Carbon-14	3,100
Cesium-134 + Cesium-137	0.18
Cobalt-60	0.0000019
Plutonium	0.0077
Strontium-90 + Yttrium-90	0.058
Krypton-85	500,000
Antimony-125	16
Iodine-129 + Iodine-131	0.44
Others	0.21

Source: DOE, 1995b

4.3.6.6.3 Water Quality

Savannah River Site

DOE has analyzed the expected liquid radiological releases from the Savannah River Site chemical separations facilities in the Interim Management of Nuclear Materials EIS (DOE, 1995a). These releases are presented in Table 4-46 (Grainger, 1995). The health effects from these liquid releases are discussed in Section 4.3.6.6.4 below.

Idaho National Engineering Laboratory

Chemical separation activities at the Idaho National Engineering Laboratory would not affect water quality because the facility designs would prevent any accidental or incident-free discharge of liquid effluents (DOE, 1995c).

Table 4-46 Annual Incident-Free Liquid Radiological Releases at the Savannah River Site

Element	*Ci/yr*
Tritium	1.29
Strontium-89/90	0.013
Ruthenium-103/106	0.012
Cesium-137	0.033
Promethium-147	0.045

4.3.6.6.4 Occupational and Public Health and Safety

Potential exposures to workers and the public due to chemical separation activities were analyzed at both the Savannah River Site and the Idaho National Engineering Laboratory (DOE, 1995c). To estimate health effects, this analysis defined three receptor groups:

- onsite workers assigned to operations involving spent nuclear fuel,
- 1994 offsite population residing within an 80-km (50-mi) radius of the chemical separation facilities (exposure via air), and
- offsite population whom management site surface-water emissions could affect.

Each of these three receptor groups would receive an annual maximum individual dose and an annual population dose. The maximally exposed worker dose would be limited by regulation to 5,000 mrem per year, as in the basic implementation of Management Alternative 1.

Savannah River Site

Incident-Free Impacts at the Savannah River Site

The highest estimated incident-free dose rates for conventional chemical separation operations at the Savannah River Site are presented in Table 4-47 (DOE, 1995a). These chemical separation operations could include activities related to blending the separated HEU down to LEU and converting all LEU into an oxide suitable for long-term storage. Values in Table 4-47 represent the estimated dose rates due to these activities, including actual chemical separation, blending down, and conversion to oxide. Multiplying these values by the estimated program duration of 13 years yields the doses presented in Table 4-48. These doses are converted into risks of LCF by applying the appropriate conversion factors and these results are also presented in Table 4-48. If the HEU were not blended down, but rather converted directly to oxide, the worker population dose would be higher because the conversion to oxide would take place in the Uranium Solidification Facility. In this facility, the workers would be closer to the uranium.

Table 4-47 Incident-Free Radiation Dose Rates Due to Chemical Separation at the Savannah River Site

	Maximum Individual Dose Rate (mrem/yr)	*Population Dose Rate (person-rem/yr)*
Public		
Via Air	0.66	27
Via Water	0.0098	0.033
Workers	5,000[a]	21

[a] *Assumed to be equal to the regulatory limit*

Table 4-48 Radiological Health Impacts Due to Incident-Free Chemical Separation Operations at the Savannah River Site

	Maximum Individual Dose (mrem)	*Maximum Individual Risk (LCF)*	*Population Dose (person-rem)*	*Population Risk (LCF)*
Public				
Via Air	8.6	0.0000043	351	0.18
Via Water	0.13	6.4×10^{-8}	0.43	0.00021
Workers	65,000	0.026	273	0.11

These risks must be combined with the risks of receiving/unloading the casks. Risks to the public were presented earlier in this chapter in Table 4-8. The risks of storage at RBOF are also presented, but they are very low compared to those of receipt/unloading. Assuming the foreign research reactor spent nuclear fuel would be received at RBOF for the full 13 years, the public MEI and population risks would be 7.1×10^{-10} LCF and 0.000036 LCF, respectively. These risks are much lower than the corresponding values in Table 4-48.

The handling-related risks to workers were presented earlier in this chapter. Under the conservative assumptions in the basic implementation of Management Alternative 1, the maximally exposed worker risk due to handling could be as high as 0.026 LCF which is equal to the 0.026 LCF in Table 4-48.

For the public, the estimated MEI risk from incident-free chemical separation activities would be

Table 4-49 Annual Impacts of Chemical Separation Accidents at the Savannah River Site

	Accident Frequency (per year)	Consequences (LCF)		Risks (LCF/yr)	
		Maximum Individual	Population	Maximum Individual	Population
Unpropagated Fire					
• Public	0.02	0.00018	1.3	0.0000036	0.026
• Workers	0.02	0.00086	---	0.000017	---

Table 4-50 Impacts of Accidents During Chemical Separation Operations at the Savannah River Site

	Maximum Individual Risk (LCF)	Population Risk (LCF)
Public	0.000047	0.34
Workers	0.00022	---

These results indicate that the estimated public MEI risk due to the chemical separation accidents is 0.000047 LCF. The estimated public population risk due to chemical separation accidents is 0.34 LCF. These risks must be combined with the risks of receiving/unloading and temporarily storing the foreign research reactor spent nuclear fuel, which were presented in Table 4-24. Assuming the foreign research reactor spent nuclear fuel would be received/unloaded and stored at RBOF for 13 years, the public MEI and population risks would be 0.0000026 LCF and 0.096 LCF, respectively.

The maximum of the two estimated accident-related MEI risks is 0.000047 LCF. This means that this hypothetical individual would have an additional chance of incurring an LCF of less than one in ten thousand.

The sum of the two population risks is 0.43 LCF. This means there would be an approximately 43 percent chance that one additional LCF would occur in the public population near the Savannah River Site due to accident conditions.

Idaho National Engineering Laboratory

Incident-Free Impacts at Idaho National Engineering Laboratory

The incident-free radiation dose rates for chemical separation at the Idaho National Engineering Laboratory are presented in Table 4-51 (DOE, 1995c). Multiplying these values by the estimated program duration of 12 years yields the doses presented in Table 4-52. These doses are converted into risks of LCF by applying the appropriate conversion factors and these results are also presented in Table 4-52.

Table 4-51 Incident-Free Radiation Dose Rates due to Chemical Separation at the Idaho National Engineering Laboratory

	Maximum Individual Dose Rate (mrem/yr)	Population Dose Rate (person-rem/yr)
Public		
Via Air	0.048	0.39
Via Water	0.0	0.0
Workers	5,000[a]	18

[a] *Assumed to be equal to the regulatory limit*

Table 4-52 Radiological Health Impacts Due to Incident-Free Chemical Separation Operations at the Idaho National Engineering Laboratory

	Maximum Individual Dose (mrem)	*Maximum Individual Risk (LCF)*	*Population Dose (person-rem)*	*Population Risk (LCF)*
Public				
Via Air	0.58	2.9×10^{-7}	4.7	0.0024
Via Water	0.0	0.0	0.0	0.0
Workers	60,000	0.024	216	0.086

These risks must be combined with the risks of receiving/unloading the casks. Risks to the public were presented earlier in this chapter in Table 4-9. The risks of storage are also presented, but they are very low compared to those of receipt/unloading. Assuming the foreign research reactor spent nuclear fuel would be received at FAST for the full 13 years, the public MEI and population risks would be 2.5×10^{-9} LCF and 0.000021 LCF, respectively. These risks are much lower than the corresponding values in Table 4-52.

The handling-related risks to workers were presented earlier in this chapter. Under the conservative assumptions in the basic implementation of Management Alternative 1, the maximally exposed worker risk due to handling could be as high as 0.026 LCF which is higher than the 0.024 LCF in Table 4-52.

Table 4-54 Impacts of Accidents During Chemical Separation Operations at the Idaho National Engineering Laboratory

	Maximum Individual Risk (LCF)	Population Risk (LCF)
Public	3.0×10^{-7}	0.000034
Workers	0.000044	---

The highest estimated public MEI risk is 3.0×10^{-7} LCF, which means that an individual living at the management site boundary would have an additional chance of incurring an LCF of less than one in a million.

The highest estimated public population risk is 0.000034 LCF, which is much less than one LCF.

4.3.6.6.5 Waste Management

Savannah River Site

DOE has analyzed the wastes that would be generated from the aluminum-based foreign research reactor spent nuclear fuel and from an additional inventory of aluminum-based spent nuclear fuel during chemical separation activities. High-level waste, saltstone, transuranic waste, hazardous/mixed waste, and low-level waste would be generated under this implementation subalternative. All these wastes would be managed along with similar wastes at the Savannah River Site. The alternatives for managing all DOE wastes have been evaluated in the Draft Waste Management Programmatic EIS (DOE, 1995b).

The estimates of waste volumes that would be generated under this implementation subalternative are based on comparisons with similar operations on similar spent nuclear fuels. These aluminum-based spent nuclear fuel elements are similar to DOE's Mark 16/22 spent nuclear fuel elements at the Savannah River Site.

High-level liquid waste would be transferred to the F/H-Area Tank Farm for volume reduction and then to the Defense Waste Processing Facility for conversion into a borosilicate glass form suitable for prolonged storage. The high-level glass waste that would result from chemically separating the 18.2 MTHM (approximately 17,800 elements) of foreign research reactor spent nuclear fuel in this implementation subalternative would fill about 72 canisters (Dupont, 1996). Scaling this result up to include the total inventory of 51 MTHM yields an estimate of about 200 canisters. These canisters would be managed with the estimated 5,717 canisters that the Savannah River Site expects to produce from the existing onsite inventory of liquid high-level waste (WSRC, 1995). Each canister will contain approximately 40,000 curies of radioactivity (DOE, 1994a). The representative radionuclide composition of the waste glass is presented in Table 2.11 of the Integrated Data Base Report-1993 (DOE, 1994a). The radionuclides that contribute most of the radioactivity would be cesium-137, strontium-90, and their daughters. DOE expects that this waste form would be acceptable for disposal in a geologic repository.

Saltstone would be produced during the vitrification of high-level waste at the Defense Waste Processing Facility. An estimated 4,000 m^3 (140,000 ft^3) would be generated from the 18.2 MTHM of foreign research reactor spent nuclear fuel and would be disposed of onsite (Dupont, 1996). Scaling this result up to include the total inventory of 51 MTHM yields an estimate of about 11,300 m^3 (400,000 ft^3). This is much less than the maximum estimated cumulative saltstone to be generated at the Savannah River Site during the 10-year period from 1995 through 2004, which would be 625,211 m^3 (about 22,000,000 ft^3) (DOE, 1994b). The saltstone would contain far less radioactivity than the high-level waste glass: approximately 0.1 curie per cubic meter (DOE, 1994a). The approximate composition of the saltstone in terms of specific radionuclides is presented in Table C.5 of the Integrated Data Base Report-1993 (DOE, 1994a). The radionuclides that contribute most of the radioactivity would be promethium-147 until about 2000, then strontium-90 and its daughter thereafter.

Transuranic waste would not be generated during the chemical separation activities of foreign research reactor spent nuclear fuel (DOE, 1995a). The trace amounts of transuranic elements would not be removed from the waste stream, so they would be included in the high-level waste. If the Taiwan Research Reactor spent nuclear fuel (included in the total inventory of 51 MTHM) is chemically separated and the transuranic elements removed, then an estimated 832 m^3 (about 29,400 ft^3) of transuranic waste [illegible]

waste to be generated at the Savannah River Site during the 10-year period from 1995 through 2004, which would be 9,426 m^3 (about 333,000 ft^3) (DOE, 1994b).

Hazardous/mixed waste would also be produced under this implementation subalternative. An estimated 104 m^3 (about 3,700 ft^3) would be generated during 13 years of chemical separation operations (DOE, 1995a). This is much less than the maximum estimated cumulative mixed waste to be generated throughout the entire Savannah River Site during the 10-year period from 1995 through 2004, which would be 14,720 m^3 (about 520,000 ft^3) (DOE, 1994b).

Solid low-level waste would also be produced under this implementation subalternative. An estimated 74,000 m^3 (about 2,600,000 ft^3) would be generated during 13 years of chemical separation operations

Another possibility exists if large quantities of nonaluminum-based spent nuclear fuels are being chemically separated in these facilities. Some aluminum is necessary to produce the stable waste form, and the 18.2 MTHM of aluminum-based foreign research reactor spent nuclear fuel could satisfy this requirement. In this case, the chemical separation of the aluminum-based spent nuclear fuel would not increase the number of canisters that would be generated at the Idaho National Engineering Laboratory.

The estimates of low-level waste grout that would be generated under this implementation subalternative are also based on comparisons with similar operations on similar spent nuclear fuels. For 12 MTHM of aluminum-based foreign research reactor spent nuclear fuel, 1,280 m^3 (about 45,000 ft^3) of low-level waste grout would be generated (Denney, 1995). Scaling up to the 19.2 MTHM of foreign research reactor spent nuclear fuel in this implementation subalternative yields about 2,000 m^3 (70,629 ft^3). Scaling up further to the total inventory of 65 MTHM yields an estimate of about 6,900 m^3 (about 245,000 ft^3). This grout would be managed along with the other grout the Idaho National Engineering Laboratory would produce onsite. The grout is expected to contain far less radioactivity than the high-level waste glass/ceramic: much less than one curie per cubic meter. The composition of the grout in terms of all the specific radionuclides has not been determined yet, but the major radioactive constituents would be cesium-137 and strontium-90. The cesium-137 and strontium-90 concentrations in the grout are expected to be about 0.034 and 0.0093 curies per cubic meter, respectively (Bendixsen, 1995).

Transuranic waste would not be generated during chemical separation of the foreign research reactor spent nuclear fuel. Furthermore, the Idaho National Engineering Laboratory would not separate the transuranic elements from the Taiwan Research Reactor spent nuclear fuel if it were transported there from the Savannah River Site. Therefore, no transuranic waste would be generated during chemical separation of the additional inventory of spent nuclear fuel. The estimated amount of cumulative transuranic waste for 10 years with minimum waste management at the Idaho National Engineering Laboratory is 67,000 m^3 (about 2,400,000 ft^3) (DOE, 1995c).

Hazardous/mixed waste would also be produced under this implementation subalternative. Assuming a waste generation rate about equal to the rate at the Savannah River Site, an estimated 96 m^3 (3,400 ft^3) would be generated during 12 years of chemical separation operations. This is much less than the estimated 29,000 m^3 (1,020,000 ft^3) of cumulative hazardous and mixed waste to be generated throughout the entire Idaho National Engineering Laboratory during the next 10 years with minimum waste management (DOE, 1995c).

Solid low-level waste would also be produced under this implementation subalternative. Assuming a waste generation rate about equal to the rate at the Savannah River Site, an estimated 68,300 m^3 (2,400,000 ft^3) would be generated during 12 years of chemical separation operations and would be disposed of onsite. This is more than the estimated 47,000 m^3 (1,660,000 ft^3) of low-level waste to be generated throughout the entire Idaho National Engineering Laboratory during the next 10 years with minimum waste management (DOE, 1995c). The Idaho National Engineering Laboratory would treat the waste at the Waste Experimental Reduction Facility and send it to the Radioactive Waste Management Complex for onsite disposal.

4.3.6.7 Summary of the Impacts of Implementation Alternative 6 (Near Term Conventional Chemical Separation)

The principal impacts under this implementation alternative would be occupational and public health and safety impacts. These are presented in Table 4-55 in terms of the risk of death due to cancer during each of the four segments of this implementation alternative. It also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. The marine transport, port activities, and ground transport impacts are identical to the basic implementation of Management Alternative 1. The management site activity impacts were derived by comparing, and summing as appropriate, the handling impacts of the basic implementation of Management Alternative 1 and the impacts of chemical separation dedicated to foreign research reactor spent nuclear fuel. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-55 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE site receiving the maximum dose allowed by DOE regulation (5,000 mrem) every year.

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have approximately a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 0.0000043 LCF.

The highest estimated accident MEI risk is 0.000047 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than one in ten thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

The population risks were calculated by summing the appropriate spent nuclear fuel handling risks from the basic implementation of Management Alternative 1 with the risks of chemical separation at each management site and selecting the largest value. For example, the incident-free worker population risk of 0.21 LCF is the largest sum of the risks from that estimated for spent nuclear fuel handling operations under Phase 1 of the basic implementation of Management Alternative 1 and the estimated risk due to chemical separation dedicated to foreign research reactor spent nuclear fuel at the Savannah River Site or the Idaho National Engineering Laboratory. The sum of the above risks at the Idaho National Engineering Laboratory is 0.19 LCF [0.10 LCF from Phase 1 of the basic implementation (Table 4-15) and 0.086 LCF from chemical separation], and the corresponding value at the Savannah River Site is 0.21 LCF [0.10 LCF from Phase 1 of the basic implementation (Table 4-14) and 0.11 LCF from chemical separation].

As shown in Table 4-55, the total incident-free population risk would be 0.39 LCF for the potentially exposed public, while the corresponding risk would be 0.32 LCF for workers. Thus, there would be an estimated 39 percent chance of incurring 1 additional LCF among the exposed general public, and a 32 percent chance of incurring 1 additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-55. DOE and the Department of State estimate there could be about a 13 percent chance that a truck driver or member of the public could die in a traffic accident associated with this implementation alternative. This death would be unrelated to the radioactive nature of the cargo.

Table 4-55 Maximum Estimated Radiological Health Impacts of Implementation Alternative 6 (Near Term Conventional Chemical Separation)

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	Population	
		General Public	Workers
Marine Transport			
Incident-Free	0.00052	0	0.034
Accidents	5×10^{-10}	much less than 0.000029	---
Port Activities			
Incident-Free	0.00052	0	0.012
Accidents	2×10^{-10}	0.000029	---
Ground Transport			
Incident-Free	0.00052	0.21	0.065
Accidents	1.3×10^{-11}	0.00014	---
Site Activities			
Incident-Free	0.026	0.18	0.21
Accidents	0.000047	0.43	---
Highest Individual Risk			
Incident-Free	0.026	----	---
Accidents	0.000047	----	----
Total Population Risk			
Incident-Free	----	0.39	0.32
Accidents	----	0.43	----

4.3.7 Implementation Alternative 7: New Developmental Treatment and/or Packaging Technologies

The environmental impacts of the developmental treatment and/or packaging technologies cannot be estimated with confidence at this time because the technologies and procedures are still under development. Implementation of certain of these technologies would require new facilities and thus would generate all the impacts associated with construction. Appropriate NEPA documentation would be prepared to support a decision on implementation of a new technology. The developmental treatment and/or packaging technologies are described in Chapter 2, Section 2.2.2.7.

The date at which a new facility would be operational is highly uncertain. A fairly simple technology implemented in existing facilities could be operational by 2000. On the other hand, the technology development, NEPA analysis, facility construction, and startup could take about 15 years for a complex technology. Thus, DOE could choose to implement one of the accept-and-store alternatives, in parallel with this alternative to prepare the foreign research reactor spent nuclear fuel for disposal. This may be necessary because foreign research reactor spent nuclear fuel may not be accepted in a geologic repository without some form of chemical processing or treatment. The repository acceptance criteria will not be final until a repository has been licensed.

Any new facilities would be designed to meet modern standards. The new design would minimize air and water emissions and the public and worker radiation doses at least as well as existing facilities, so DOE and the Department of State expect these impacts would be somewhat lower than those presented above for the conventional chemical separation technologies.

Some rough quantitative estimates are possible on the number of canisters that would be produced by some of the developmental technologies for disposal. Table 4-56 compares these estimates to the number of canisters that would be generated by chemical separation. The estimates of numbers of canisters that would be generated by the developmental treatment and/or packaging technologies do not depend on which DOE site performs the treatment and/or packaging.

Table 4-56 Comparison of Geologic Disposal Canisters for Various Technologies

Technology	Approximate Number of Canisters
Conventional Chemical Separation	
at the Savannah River Site	72
at the Idaho National Engineering Laboratory	90
Developmental Packaging Technologies	
Direct Disposal in Small Packages	140
Can-in-Canister	240
Developmental Treatment Technologies	
Melt and Poison	25
Chop and Poison	25
Melt and Dilute	180
Dissolve and Poison	950
Chop and Dilute	4,900
Dissolve and Dilute	11,800

The can-in-canister concept was recently introduced (Leventhal and Lyman, 1995), but it could be possible to implement it quickly at the Savannah River Site. Most of the foreign research reactor spent nuclear fuel elements would fit in cans of approximately 10 cm diameter and 85 cm length. If all of the approximately 22,700 elements were placed in these cans, the total canned volume would be about 150 m^3. Using the can-in-canister technology, this volume of glass would be displaced from high-level waste canisters to be produced in the Defense Waste Processing Facility. Since each canister has an internal volume of 0.625 m^3, displacing 150 m^3 of glass would require the production of approximately 240 additional high-level waste glass canisters at the Defense Waste Processing Facility.

The rest of the estimates of numbers of canisters that would be generated by the developmental technologies are scaled from a study (WSRC, 1994a) of the disposition of 7.3 MTHM of aluminum-based spent nuclear fuel, up to the 18.2 MTHM of aluminum-based foreign research reactor spent nuclear fuel. The melt and poison or chop and poison technologies could produce the fewest canisters, as low as 25 canisters. The consolidate and poison technology could produce the next lowest number of canisters (about 140) among the developmental technologies analyzed. The can-in-canister, melt and dilute, dissolve and poison, chop and dilute, and dissolve and dilute technologies would produce increasing numbers of canisters, in that order. The most canisters would be produced by the dissolve and dilute technology: over 11,000 canisters. This uncertainty in the number of canisters translates into a large uncertainty in the cost of disposal. Furthermore, it is not clear which, if any, of these waste forms would be acceptable in a geologic repository.

4.4 Management Alternative 2: Facilitate the Management of Foreign Research Reactor Spent Nuclear Fuel Overseas

The basic implementation of Management Alternative 1 of the proposed action and the seven implementation alternatives to the basic implementation of Management Alternative 1 are all based on acceptance of foreign research reactor spent nuclear fuel into the United States. As discussed in Chapter 2, the two subalternatives under Management Alternative 2 facilitate overseas management of foreign research reactor spent nuclear fuel. This section discusses their policy considerations and environmental impacts. For convenience, the two subalternatives under Management Alternative 2 are defined briefly below:

1. Subalternative 1a - Overseas storage of the foreign research reactor spent nuclear fuel with U.S. technical and/or financial assistance, and

2. Subalternative 1b - Overseas reprocessing of the foreign research reactor spent nuclear fuel with U.S. nontechnical assistance.

Under these subalternatives, no foreign research reactor spent nuclear fuel would be accepted into the United States. The United States would negotiate some form of technical assistance and/or financial incentives in return for maintaining some measure of control over the spent nuclear fuel containing uranium enriched in the United States.

4.4.1 Subalternative 1a: Overseas Storage with U.S. Assistance

Policy Considerations

The foreign research reactor spent nuclear fuel could remain in interim storage overseas. The number of foreign research reactor spent nuclear fuel management sites involved would be greater and the quality of storage technology in some countries might be lower than if the basic implementation of Management Alternative 1, or one of its seven implementation alternatives, was adopted.

The cost of this subalternative might be greater than the cost of the basic implementation of Management Alternative 1 because it might not take advantage of economies of scale. To set up a secure area and a nuclear material handling infrastructure, purchase a storage cask, transfer the spent nuclear fuel to the cask, and maintain the secure area and nuclear infrastructure for 40 years would cost tens of millions of dollars. To repeat this in several dozen countries could potentially push the total cost up into the range of hundreds of millions of dollars. Furthermore, after incurring this expense, all of the U.S. origin HEU would still be located in foreign countries where a change in government could reverse any commitment to withhold the material from production of nuclear weapons.

This subalternative would be economically attractive only in countries that already have nuclear infrastructures. In these cases, the addition of the spent nuclear fuel from research reactors to existing spent nuclear fuel inventories in storage would involve only incremental costs without all the startup costs.

If the United States does not accept any near term foreign research reactor spent nuclear fuel shipments, provision of U.S. technical and/or financial assistance for the development of safe and secure storage capabilities would help to alleviate some of the problems posed by a lack of sufficient storage capacity. However, this subalternative presents several drawbacks from a nuclear weapons nonproliferation policy standpoint. The accumulation overseas of ever larger amounts of spent nuclear fuel containing HEU poses a risk that such weapons-usable material might be illicitly diverted to a weapons program. Although U.S. assistance in maintaining adequate physical security for foreign research reactor spent nuclear fuel repositories may lessen the potential for diversion, the proliferation risks would still be greater than under the basic implementation of Management Alternative 1. As the foreign research reactor spent nuclear fuel ages, it would become less radioactive and thus a more attractive target for illicit diversion.

For countries that will not allow the indefinite storage in their territories of increasing quantities of spent nuclear fuel, this subalternative is not a viable option. Under this scenario, reactor operators in these countries, in order to avoid shutting down, might be forced to consider storing their spent nuclear fuel in other countries, where safe and secure management and material accountancy problems could exist and the risk of illicit diversion could be a concern. For example, Austria was reportedly approached by commercial interests from Belarus with an offer to store spent nuclear fuel from the ASTRA reactor for hard currency. (Since the "Offsite Fuels Policy" for HEU spent nuclear fuel expired in 1988, the Austrian government has required that for fresh fuel to enter the country, an equivalent quantity of spent nuclear fuel must be shipped out of the country.) The offer, which was rejected in support of nuclear weapons nonproliferation policies, is indicative of the scenarios that may develop as pressure builds on reactor operators to close the back end of their nuclear fuel cycle.

Impacts

There would be no environmental impacts on U.S. territory for the duration of the interim period.

4.4.2 Subalternative 1b: Overseas Reprocessing with United States Non-Technical Assistance

4.4.2.1 Overview and Policy Considerations

Foreign research reactor spent nuclear fuel could be reprocessed in foreign facilities and the resulting high-level waste vitrified or cemented. No U.S. reprocessing technology would be used in this subalternative. The inventory and conditions for management of foreign research reactor spent nuclear fuel under Subalternative 1b are the same as those under basic implementation of Management Alternative 1. The amount of HEU that would be removed from international commerce is the same as under basic implementation of Management Alternative 1 [4.6 metric tons (5.1 tons)]. To be consistent with U.S. nuclear weapons nonproliferation policy, however, bilateral agreements would have to be established with one or more foreign governments before DOE and the Department of State could consider implementation of such a subalternative.

The advantages and disadvantages of the technology used for reprocessing overseas would be essentially the same as those described for chemical separation in the United States as discussed in Section 2.2.2.6.

There are four sites in Europe at which reprocessing is conducted for commercial customers: the Marcoule and La Hague sites in France, and the Dounreay and Sellafield sites in the United Kingdom. The companies that operate these sites are strictly regulated by their government agencies. The facilities at La Hague and Sellafield are dedicated to oxide spent nuclear fuel from commercial reactors and are not likely candidates for reprocessing the metallic foreign research reactor spent nuclear fuel. All four of these sites routinely release small quantities of radionuclides into the environment and produce radioactive wastes. For example, in 1993 the releases from the Dounreay facility to the North Sea included 2.7 Ci of total alpha activity, 220 Ci of beta activity excluding tritium, and 27 Ci of beta activity from tritium. These releases represented 13 percent, 7.2 percent, and 0.8 percent of the applicable regulatory limits (Jones et al., 1994). The radionuclides released into the atmosphere and into a river or sea would flow across international boundaries. These releases would cause a small, unmeasurable increase in world-wide natural background radiation levels. The transport of vitrified high-level waste away from the reprocessing facility would also produce environmental impacts on foreign territory and possibly in international waters.

Since the United States does not encourage the development of reprocessing capabilities overseas, DOE and the Department of State would only consider this subalternative in France or the United Kingdom where the capability already exists. Reprocessing would most likely take place (as it already has in several instances) at the Dounreay facility—the sole facility currently willing and able to reprocess foreign research reactor spent nuclear fuel. France's facility in Marcoule does reprocess spent nuclear fuel from French research reactors, but does not currently accept such spent nuclear fuel from other nations for reprocessing.

The British and French regulatory agencies require the customer to accept the wastes as a condition of reprocessing spent nuclear fuel, so this option would be unavailable to those countries lacking the technical or legal capability to store or dispose of high-level waste. Alternatively, the United States might consider accepting the wastes from reprocessing.

4.4.2.2 Waste Generation at the Foreign Reprocessing Site

Reprocessing the foreign research reactor spent nuclear fuel would produce two distinct streams: the uranium and the waste products.

For spent nuclear fuel containing HEU, the HEU would be blended down to LEU at the reprocessing facility. If the LEU were then shipped to the United States, the resulting environmental impacts would be no greater than for ordinary nonhazardous cargo because LEU produces such a small radiation dose rate.

The British and French have decades of experience in conditioning nuclear waste at their four reprocessing facilities. In recent years, they have greatly reduced the volumes of wastes that require disposal. Both nations use the same technology for vitrifying their high-level waste, and both nations produce the same size high-level waste glass canister: 0.15 m^3 (5.3 ft^3). These canisters of high-level waste glass are expected to be suitable for disposal in geologic repositories. As of September 1993, France and the United Kingdom had filled more than 2,100 and 350 canisters with high-level waste glass, respectively (Masson, et al., 1994).

As a general rule, European reprocessing and vitrification of about 8 to 10 MTHM of spent nuclear fuel would generate about 1 m^3 (35.3 ft^3) of high-level waste in glass form (UKAEA, 1994; Masson, et al., 1994). Thus, if all 19.2 MTHM of the foreign research reactor spent nuclear fuel were reprocessed and vitrified overseas, DOE and the Department of State estimate that the total volume of vitrified high-level waste would be only about 2.4 m^3 (85 ft^3). DOE and the Department of State estimate that the high-level waste from reprocessing all the foreign research reactor spent nuclear fuel would fill about 16 European-sized canisters. For reference, this volume of glass waste would fill four American-sized canisters.

4.4.2.3 Removal of Waste from the Reprocessing Site(s)

The British and French governments do not accept responsibility for ultimate disposal of the high-level waste glass canisters for foreign customers. Both nations require that disposal of the high-level waste glass canisters and any other wastes generated during reprocessing of their spent nuclear fuel, including low-level waste, be the responsibility of the nation(s) hosting the reactors. At the Dounreay Site, however, only small amounts of low-level waste have been generated during reprocessing of spent nuclear fuel from research reactors. Many nations with foreign research reactors, however, do not have any capabilities to accept the high-level waste glass canisters. The United States may accept the intact foreign research reactor spent nuclear fuel from these nations while simultaneously encouraging the nations which can accept the canisters to reprocess their foreign research reactor spent nuclear fuel under the conditions noted in Section 4.4.2.1. This would be a combination of the basic implementation of Management Alternative 1 and Subalternative 1b (overseas reprocessing) of Management Alternative 2.

As another option under this subalternative, if the host nations cannot accept this responsibility, the United States would commit to accept the high-level waste glass canisters. This could provide the incentive necessary to convince reactor operators to cooperate with the RERTR program and to use LEU in their reactors. Some nations may refuse to reprocess or require the United States to take title to the foreign research reactor spent nuclear fuel prior to reprocessing.

DOE and the Department of State could begin accepting canisters into the United States within the first 10 years, or DOE and the Department of State could specify that they be stored at the reprocessing facility for decades. If the canisters were accepted in the near term, they would most likely be stored at the Savannah River Site because this site has already built a new storage facility with a capacity of 2,286 canisters. If the canisters were stored overseas for decades, then they would be transported directly to the geologic repository.

Marcoule produces vitrified waste, similar to U.S. vitrified waste. In the United Kingdom on the other hand, as a result of a different regulatory structure, the wastes from reprocessing of research reactor spent nuclear fuel are classified as intermediate-level radioactive wastes. (In the United States, these same materials would be classified as high-level radioactive wastes.) In the United Kingdom, the intermediate-level wastes are mixed with a special cement and poured into steel drums, which can then be buried. This waste form is dissimilar to the vitrified borosilicate glass high-level waste form that is expected to be produced in the United States, and is incompatible with United States radioactive waste disposal standards. The government of the United Kingdom might allow an exchange of vitrified commercial waste from Sellafield for cemented waste from Dounreay, which might allow the United States to accept vitrified high-level waste from the United Kingdom.

Transportation of vitrified high-level waste must conform to U.S. Department of Transportation (49 CFR Part 173) and NRC (10 CFR 71) regulations. Under this option, the European-sized glass canisters would be transported in "Type B" casks, which provide a high degree of assurance that cask integrity will be maintained with essentially no loss of radioactive contents or serious impairment of the shielding capability provided by the cask, even in severe accidents. DOE has prepared initial designs for a defense high-level waste cask for truck transportation of the Savannah River Site high-level waste. As initially designed, the defense high-level waste cask uses a solid body concept to absorb energy during an accident and normal transportation conditions. To minimize the exposure to gamma radiation, shielding would be provided by a depleted uranium liner inside the cask body. (Gamma radiation is high-energy, short wavelength electromagnetic radiation with properties similar to x-rays.) The regulatory limit for radiation dose rate outside the cask is 10 mrem per hour at 2 m (6.6 ft) from the edge of the vehicle. Casks transported under this option are assumed to emit this level of radiation. Currently, however, no casks for shipping high-level waste canisters by truck or rail have been certified by the NRC.

Each of these "Type B" casks would be large enough to hold two European-sized glass canisters. Thus, the option of overseas reprocessing with acceptance of approximately 16 high-level waste glass canisters would require about 8 cask shipments into the United States (versus 721 cask shipments by sea and 116 by land under the basic implementation of Management Alternative 1). Vitrified high-level waste shipments would use the same East Coast port(s) identified in Chapter 2 for foreign research reactor spent nuclear fuel. The same procedures and representative overland routes analyzed for foreign research reactor spent nuclear fuel would apply to these shipments of vitrified high-level waste. The management site for these canisters would be the Savannah River Site. Alternatively, they might be transported directly to the candidate geologic repository at Yucca Mountain, NV.

Each of the eight casks is assumed to contain the waste products associated with one-eighth of the foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1.

Marine Transport Impacts

Risks under Subalternative 1b were assessed using the same methodology used to evaluate risks associated with the transport of the foreign research reactor spent nuclear fuel. The major differences in the analysis are the number of cask shipments and the isotopic content within each transportation cask.

Impacts of Incident-Free Marine Transport

As with the shipment of foreign research reactor spent nuclear fuel, the primary impact of incident-free marine shipping of vitrified waste would be upon the crews of the ships. Most of the assumptions used in the analysis of the crew exposure to the spent nuclear fuel (see Section 4.2.2.2) have been used to analyze the impact of the shipment of vitrified waste. The primary contribution to the crew dose would come from the daily cargo inspection activities. Three crew members have been modeled as performing the inspections and the same three crew members are assumed to perform this task for the entire voyage. For the purposes of this analysis it has been assumed that the vitrified waste would be transported on a chartered vessel, there would be no intermediate port calls, and the shipment would originate in Europe (either the United Kingdom or France.)

As in the spent nuclear fuel analysis, either two or eight casks are assumed to be on each single voyage. This assumption results in exposure to two radiation fields during all activities that bring crew members into the vicinity of the transportation casks. Should all the casks be shipped at once, this assumption is equivalent to assuming that this single voyage is made with two casks per hold in one vessel. The crew risk would be the same for this single voyage as for four voyages with two casks per vessel.

Results of the marine incident-free risk analysis are presented in Table 4-57. Due to the reduced number of cask shipments, compared to the approximately 721 marine cask shipments of foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1, the risks to the crew would be approximately 2 orders of magnitude lower than those calculated in Section 4.2.2.2 for the basic implementation of Management Alternative 1. The doses to the crew, including the maximally exposed worker, would be well below the DOE and NRC limits for public exposure of 100 mrem per year. If, however, all the casks were shipped in 1 year (perhaps all on one ship), then the maximally exposed worker dose would exceed the limit of 100 mrem per year. In this case, new inspectors would be used to keep each individual's dose below the limit.

Table 4-57 Incident-Free Marine Transport Impacts (Subalternative 1b)

	Maximally Exposed Worker Dose (mrem)	*Maximally Exposed Worker Risk (LCF)*	*Population Dose to Crew (person-rem)*	*Population Crew Risk (LCF)*
Per voyage (2 casks)	53	0.000021	0.19	0.00007
Entire program	210	0.000084	0.74	0.00030

Impacts of Accidents During Marine Transport

If the ship carrying a cask of vitrified waste were to catch fire at sea and the cask was sufficiently damaged by fire to release its contents, members of the ship's crew near the fire would be exposed to the released radioactive material. Any resulting plume carrying radioactive particles would disperse over the ocean, where there is no human population. Therefore, the ship's crew would be the only people exposed to the released radioactive material. The number of ship's crew members is considerably smaller than the population modeled within a short distance of an accident that occurs in the port. Therefore, consequences of a shipboard accident resulting in the release of radioactive material in a plume would be covered by the consequences of the accidents considered in the port analysis. As discussed below, because the oceans are a very dilute system, effects on marine biota would not be discernible.

If a collision or other accident (e.g., loss of a cask over the side in a storm) occurred in which an intact cask fell overboard, the fact that the cask would be immersed would not necessarily result in a release of its contents. Spent nuclear fuel casks are designed to withstand at least a 15-m (50-ft) immersion, and it has been demonstrated that the cask seals will remain intact at much greater depths (IAEA, 1990). Spent nuclear fuel casks, damaged or undamaged, can be recovered from water up to 200 m (660 ft) deep: well beyond the range typical of coastal and port depths. (Recovery at great depths, e.g., more than 2,000 m or 6,600 ft, is possible, but would be costly). It is reasonable to believe that a cask would be recovered in any incident involving the immersion of a cask in waters up to 200 m (660 ft) in depth.

The Nuclear Energy Agency of the Organization for Economic Cooperation and Development, Paris, France, estimated the impacts of various accident scenarios involving shipment of reprocessed commercial

Humans would not be the principally exposed species in a deep ocean accident involving vitrified waste casks. Using the Nuclear Energy Agency estimates and assuming that the damaged waste cask lay on the ocean floor where it slowly released its radioactive inventory, the peak doses to biota residing on the ocean floor in or near the uppermost sediment layer would be 0.9 rad per year for fish, 1.2 rad per year for crustaceans, and 41 rad per year for mollusks (NEA, 1988).

Harmful effects of chronic irradiation have not been observed in natural aquatic populations at dose rates less than 365 rad per year (NCRP, 1991). At doses an order of magnitude below this, as would be the case in an accident involving the vitrified waste from the foreign research reactor spent nuclear fuel, it is unlikely that either a population of marine biota or individual members of that population would be harmed by the radiation resulting from a spent nuclear fuel accident. Furthermore, no chemical hazard would be expected from the release of the contents of the vitrified waste canisters into the open ocean.

Using the same accident probabilities used in the marine transport analysis of the basic implementation of Management Alternative 1, risk estimates were developed for this subalternative. The MEI risk due to the loss of a vitrified high-level waste cask in the ocean is very low for the shipment of up to eight casks. The highest estimated risk to a human would occur in the accident scenario in which a cask is sunk and not recovered from coastal waters. This scenario would result in an MEI risk on the order of 1×10^{-10} mrem per year, which corresponds to about 2.7×10^{-15} LCF. This means that the chance of the MEI incurring one LCF due to this subalternative would be about one in one quadrillion.

Port Activity Impacts

Impacts of Incident-Free Port Activities

As with the shipment of the foreign research reactor spent nuclear fuel, the primary impact of incident-free port activities required to unload the vitrified waste casks is upon the workers: port handlers, inspectors, and port staging personnel. Most of the assumptions used in the analysis of the port worker exposure to the foreign research reactor spent nuclear fuel (see Section 4.2.2.2) have been used to analyze the impact of the shipment of vitrified waste.

Results of the port activities' incident-free risk analysis are presented in Table 4-58. Due to the reduced number of cask shipments, compared to the approximately 721 marine cask shipments of foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1, the risks to the port workers are approximately 2 orders of magnitude lower than those calculated in Section 4.2.2.2 for the basic implementation of Management Alternative 1. The doses to the crew, including the maximally exposed worker, are well below the DOE and NRC limits for public exposure of 100 mrem per year.

Impacts of Accidents During Port Activities

The methodology used to evaluate the accident consequences and risks associated with port accidents is identical to that used to assess these items for the shipment of foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1 (Section 4.2.2.3). The MACCS code was used with site-specific population and meteorology data to determine the consequences of an accident. The inventory (radionuclide content) of the transportation casks was determined by combining the radionuclide content of all of the vitrified waste to be returned to the United States under this subalternative and equally dividing it among the eight casks. In this analysis it was assumed that the Canadian spent nuclear fuel, which was assumed to be sent to the United States via truck in the analysis documented in Section 4.2.2.3, would be sent to Europe and reprocessed. The vitrified waste from this spent nuclear fuel is included in this analysis.

Table 4-58 Incident-Free Port Activity Impacts (Subalternative 1b)

Impacts Per Shipment				
	Maximally Exposed [illegible]	*Maximally Exposed* [illegible]	*Population Dose* [illegible]	[illegible]

Table 4-59 Port Accident Risks (Subalternative 1b)

Port	Risk per Single-Cask Shipment of Waste		Risk of the Entire Waste Acceptance Option	
	Population Dose (person-rem)	LCF	Population Dose (person-rem)	LCF
Philadelphia	0.006	0.000003	0.05	0.00002
Charleston	0.001	0.0000007	0.01	0.000005
MOTSU	0.0005	0.0000002	0.004	0.000002

The MEI doses calculated for these accidents have a rather small variance. The largest estimated MEI dose is 740 mrem. The largest probability of one LCF (given that the accident has occurred) was 0.00035. Combining these estimates with the probability of a severity category 4 accident per shipment and the number of shipments results in an MEI risk of 1.8×10^{-8} LCF.

Ground Transport Impacts

Under Subalternative 1b, DOE and the Department of State would transport eight casks of vitrified high-level waste overland from an East Coast port(s) to a candidate geologic repository (in Nevada for example). The shipments may go directly from the port(s) to the candidate geologic repository or they might go from the ports to the Savannah River Site for storage, then from the Savannah River Site to the candidate geologic repository. Results are displayed in Figures 4-18 and 4-19.

Impacts of Incident-Free Ground Transport (Ports to Repository)

Impacts of incident-free ground transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. The dose rate near vehicles carrying vitrified waste was assumed to equal the regulatory limit of 10 mrem per hour at 2 m (6.6 ft) from the vehicle. Incident-free transportation of vitrified high-level waste was estimated to result in total latent fatalities that ranged from 0.00023 to 0.0032 over the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The estimated number of radiation-related LCF for transportation workers ranged from 0.0001 to 0.0008. The estimated number of radiation-related LCF for the general population ranged from 0.00009 to 0.0024, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.0001 to 0.0005. The impacts of transportation of vitrified waste canisters are described in more detail in Appendix E.

To estimate the maximally exposed ground transport worker risk, DOE and the Department of State assumed all the vitrified waste was transported during a 1-year period and one truck driver received his annual limit of 100 mrem during that year. This dose translates into a risk of 0.00005 LCF.

Impacts of Accidents During Ground Transport (Ports to Repository)

Cumulative transportation accident risks over the vitrified waste shipment program are estimated to range from 0.0000002 to 0.0000059 LCF from radiation and from 0.00003 to 0.0016 for traffic fatality, depending on the transportation mode and the port(s) selected.

Impacts of Incident-Free Ground Transport (Ports to the Savannah River Site to Repository)

Impacts of incident-free ground transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. The dose rate from casks containing vitrified waste was assumed to equal the regulatory limit of 10 mrem per hour at 2 m (6.6 ft) from the vehicle. The

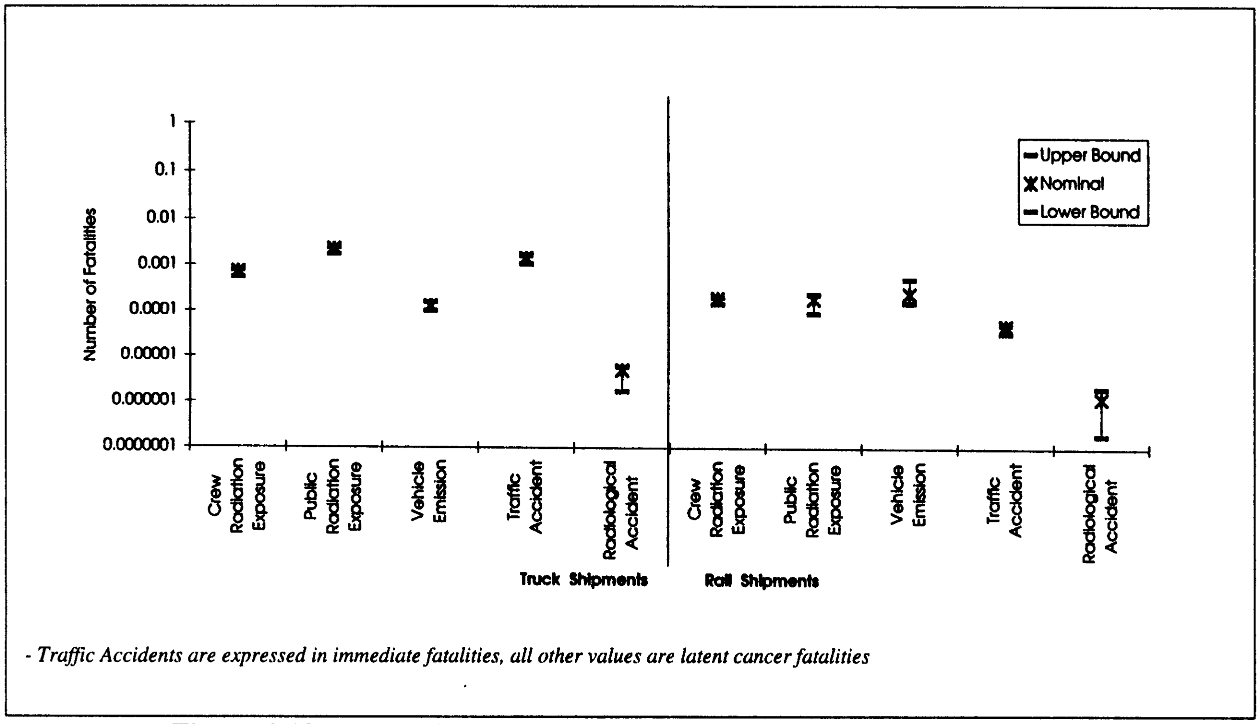


Figure 4-18 Range of Estimated Fatalities (Latent and Immediate) Under Management Alternative 2, Subalternative 1b (Ports to Repository)

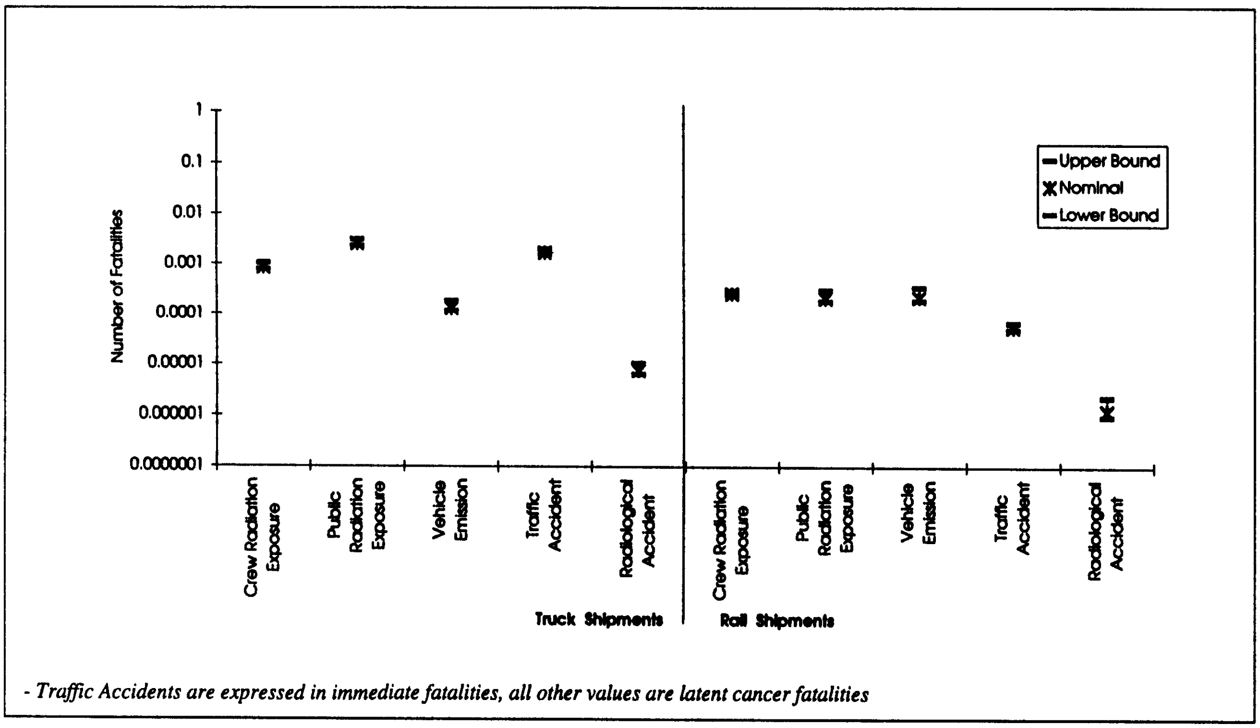


Figure 4-19 Range of Estimated Fatalities (Latent and Immediate) Under Management Alternative 2, Subalternative 1b (Ports to Savannah River Site to Repository)

incident-free transportation of the vitrified high-level waste was estimated to result in total latent fatalities that ranged from 0.00041 to 0.004 over the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The estimated number of radiation-related LCF for transportation workers ranged from 0.00023 to 0.001. The estimated number of radiation-related LCF for the general population ranged from 0.00018 to 0.003, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.00011 to 0.00035. Impacts of transportation of vitrified waste canisters are described in more detail in Appendix E.

To estimate the maximally exposed worker risk, it was assumed that the two legs of ground transport would be separated by a long storage period. That is, the second leg (transport from the Savannah River Site to the repository) would occur at least 20 years after the first leg (transport from the ports to the Savannah River Site). Thus, one individual truck driver would probably not be involved in both legs. DOE and the Department of State further assumed that each leg would last no more than 1 year, so no individual truck driver could receive more than the annual regulatory limit of 100 mrem. This translates into a maximally exposed worker risk of 0.00005 LCF.

Impacts of Accidents During Ground Transport (Ports to the Savannah River Site to Repository)

Cumulative transportation accident risks over the vitrified waste shipment program are estimated to range from 0.000001 to 0.00001 LCF from radiation and from 0.00005 to 0.002 for traffic fatality, depending on the transportation mode and the port(s) selected.

The consequences of the maximum foreseeable offsite transportation accident are greater than those of the basic implementation of Management Alternative 1. The frequency, however, is lower due to the reduced amount of ground transport. Maximum estimated MEI risk is reduced to 7×10^{-12} LCF.

Management Site Impacts

Impacts of Incident-Free Management Site Activities

Environmental impacts associated with the receipt and storage of the vitrified high-level waste canisters under Subalternative 1b are limited to the exposure of the working crew that would handle the incoming canisters at the site. The 16 canisters of vitrified waste (approximately 0.15 m^3 or 5.3 ft^3 each) would be received in 8 shipping casks and stored at the Glass Waste Storage Building at the Savannah River Site. The facility, described in Appendix F, has been designed for vitrified waste and has space for 2,286 canisters. Vitrification of all existing liquid high-level waste at the Savannah River Site is expected to produce a total of approximately 5,717 canisters. The impact of this additional amount of glass waste on the operational characteristics of the facility would be very low.

Vitrified waste would not contain any gaseous fission products, so there is no mechanism for incident-free emissions of radioactive material. Thus, impacts to the public near the Savannah River Site under this subalternative would be equal to zero.

To estimate the maximally exposed worker dose, DOE and the Department of State assumed that all the canisters would be received during one year. This is reasonable because of the small number of cask shipments. Then DOE and the Department of State conservatively assumed that one of the workers involved in handling these shipments would receive the maximum annual dose of 5,000 mrem allowed by regulation. This dose translates into an increased risk of 0.002 LCF.

The population dose to workers handling the eight casks would be 2.6 person-rem, based on the methodology presented in Appendix F, Section F.5 for unloading and storing in a vault-type dry storage structure. This translates into a worker population risk of 0.001 LCF.

Impacts of Accidents Onsite

The addition of 16 European-sized canisters to the thousands of larger American-sized canisters is expected to increase the accident risk by a very small increment, so this increase in the risk was not specifically analyzed in this EIS. The accident analysis for the Defense Waste Processing Facility has been reported in its Final EIS (DOE, 1994e).

Since vitrified waste contains no gaseous fission products, however, it is clear that the spent nuclear fuel element breach accident scenarios are not applicable to this subalternative. Thus, the aircraft-crash-with-fire scenario would present the highest risks. The highest annual estimates of MEI/NPAI and population risks under the basic implementation of Management Alternative 1 for this accident scenario are 1.2×10^{-9} LCF and 0.0000015 LCF, respectively (see Section 4.2.4.1). DOE and the Department of State consider these estimates to cover the risks for vitrified waste because the vitrified waste is designed to be much more stable than spent nuclear fuel in all accidents. Multiplying these annual estimates by the number of years the accident might occur (30 years) yields the risks for this alternative: 3.6×10^{-8} LCF for the MEI/NPAI risk and 0.000045 LCF for the population risk.

4.4.2.4 Disposal Site Impacts

Whether the vitrified high-level waste canisters were managed at the Savannah River Site or in Europe, eventually they would be transported to a geologic repository for disposal under this subalternative. Current planning for the U.S. candidate geologic repository at Yucca Mountain in Nevada indicates that acceptance of high-level waste canisters would begin early enough that the high-level waste from foreign research reactor spent nuclear fuel could be shipped to and emplaced in the repository before the end of the interim period.

Impacts due to handling European-sized canisters at the repository would be similar to the impacts due to handling American-sized canisters. After emplacement in the disposal site, no more impacts are expected to workers, the public, or the environment for at least 10,000 years because the radioactive material would be extremely unlikely to escape from the repository.

4.4.2.5 Summary of the Impacts of Subalternative 1b

The principal impacts under Subalternative 1b would be occupational and public health and safety impacts. These impacts would be due to the acceptance of vitrified high-level waste into the United States from Europe. (If no high-level waste were accepted, then there would be no impacts on U.S. territory.) These impacts are presented in Table 4-60 in terms of the risk of death due to cancer during each of the four segments of the affected environment. It also shows, in the bottom rows, the highest of the individual risks and the total population risks. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-60 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of high-level waste producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing

Table 4-60 Maximum Estimated Radiological Health Impacts of Subalternative 1b

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	*Population*	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.000084	0	0.0003
Accidents	2.7×10^{-15}	much less than 0.00002	---
Port Activities			
Incident-Free	0.000004	0	0.000036
Accidents	1.8×10^{-8}	0.00002	---
Ground Transport			
Incident-Free	0.00005	0.003	0.001
Accidents	7×10^{-12}	0.00001	---
Site Activities			
Incident-Free	0.002	0	0.001
Accidents	3.6×10^{-8}	0.000045	---
Highest Individual Risk			
Incident-Free	0.002	----	----
Accidents	3.6×10^{-8}	----	----
Total Population Risk			
Incident-Free	----	0.003	0.0027
Accidents	----	0.000075	----

people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE site receiving the maximum dose allowed by DOE regulation (5,000 mrem) during the 1 year of high-level waste acceptance.

The highest estimated incident-free individual risk is 0.002 LCF, which would apply to an onsite radiation worker. This individual would have a one in five hundred chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally worker risk. DOE estimates this risk to be very nearly zero LCF.

The maximum estimated accident MEI risk is 3.6×10^{-8} LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than one in ten million. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

The total incident-free population risk for both the general public and workers would be much less than one LCF.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-60. There is about a 0.2 percent chance that a truck driver or member of the public could die in a traffic accident associated with this subalternative. This death would be unrelated to the radioactive nature of the cargo.

4.5 Management Alternative 3 - Combination of Elements from Management Alternatives 1 and 2 (Hybrid Alternative)

As discussed in Section 2.4, DOE and the Department of State could combine implementation elements from Management Alternatives 1 and 2. Analysis of this example Hybrid Alternative does not signify its preference over other possible Hybrid Alternatives.

Under this Hybrid Alternative, DOE and the Department of State would facilitate reprocessing of the foreign research reactor spent nuclear fuel at western European reprocessing facilities (i.e., Dounreay or Marcoule), as in Management Alternative 2. It is assumed that the foreign research reactor operators in countries that can accept the reprocessing waste would agree to this arrangement. DOE would accept and manage the remaining foreign research reactor spent nuclear fuel in the United States as in Management Alternative 1. (Refer to Section 2.4 for a more detailed description of this Hybrid Alternative).

Based on the current capabilities of overseas reprocessors, and for purposes of this analysis, only aluminum-based foreign research reactor spent nuclear fuel is assumed to be considered for reprocessing; all TRIGA spent nuclear fuel is assumed to be stored in the United States.

Under the Hybrid Alternative, the aluminum-based foreign research reactor spent nuclear fuel to be managed in the United States would be chemically separated at the Savannah River Site as in Implementation Alternative 6 to Management Alternative 1 (near term chemical separation in the United States), discussed in Sections 2.2.2.6 and 4.3.6. The uranium and waste products from this chemical separation would be managed as described in Sections 2.2.2.6 and 4.3.6, and the impacts of these activities would be covered by the impacts presented in those sections. The TRIGA spent nuclear fuel would be transported to the Idaho National Engineering Laboratory where it would be stored at existing storage facilities until ultimate disposition. This distribution of the spent nuclear fuel is consistent with the Programmatic SNF&INEL Final EIS (DOE, 1995c) Regionalization by Fuel Type alternative.

The environmental impacts associated with the foreign research reactor spent nuclear fuel that would be accepted into the United States, and the policy considerations of the Hybrid Alternative, are discussed below.

Policy Considerations

Under the Hybrid Alternative, up to 5.3 MTHM and about 5,600 elements of foreign research reactor spent nuclear fuel would be reprocessed overseas. The rest of the foreign research reactor spent nuclear fuel included in the basic implementation of Management Alternative 1, up to 13.9 MTHM and about 17,100 elements, would be accepted into the United States. Overall, the same amount of HEU as in the basic implementation of Management Alternative 1 would be removed from international commerce, up to about 4.6 metric tons (5.1 tons) of HEU.

4.5.1 Marine Transport Impacts

Impacts of Incident-Free Marine Transport

Impacts of incident-free marine transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. Incident-free transportation of spent nuclear fuel was estimated to result in total LCF that ranged from 0.021 to 0.024 over the 13-year duration of the acceptance program. These fatalities are the sum of the estimated number of radiation-related LCF to the ships' crews.

The range of impacts results from the analysis of shipment of the spent nuclear fuel on regularly scheduled commercial breakbulk vessels and on chartered container vessels and would be the same as for vessels analyzed under the basic implementation of Management Alternative 1. As in the basic implementation of Management Alternative 1, the difference between the two estimates is a result of the shorter vessel journey time for chartered vessels due to the intermediate port stops associated with the regularly scheduled commercial transport of the spent nuclear fuel.

The highest estimate of the incident-free maximally exposed worker risk is the same as for the basic implementation of Management Alternative 1 (0.00052 LCF for all the shipments combined).

Impacts of Accidents During Marine Transport

Population risks due to accidents under the Hybrid Alternative would be reduced from those associated with the basic implementation of Management Alternative 1 because of the reduced amount of marine transport. As before, the population risks of accidents at sea are bounded by the risk of accidents in port.

The maximum consequences of the at-sea accidents for the Hybrid Alternative are no different than those of at-sea accidents associated with the basic implementation of Management Alternative 1. For an accident involving the loss of a transportation cask in coastal waters, the maximum exposure to an individual is estimated to be 14,000 mrem per year. DOE and the Department of State would mitigate this impact, however, by recovering the cask. Due to the reduced number of cask shipments compared to the basic implementation of Management Alternative 1, the likelihood of such an accident would also be reduced. The Hybrid Alternative would require approximately 63 percent of the number of shipments required under the basic implementation of Management Alternative 1. The highest estimated risk due to an accident during marine transport would therefore be 0.00012 mrem per year peak dose to a human from the loss of a damaged cask in the deep ocean. This corresponds to an MEI risk of about 3×10^{-10} LCF. This means that this individual would have a chance of less than one in a billion of incurring an LCF due to an accident during marine transport.

4.5.2 Port Activity Impacts

Impacts of Incident-Free Port Activities

In the analysis of the basic implementation of Management Alternative 1, the radiological impact of port activities was estimated on a per-shipment basis. The Hybrid Alternative would require about 63 percent of the number of cask shipments required under the basic implementation of Management Alternative 1. The incident-free impacts of the port activities are proportionally reduced. The estimated number of LCF associated with this alternative range from 0.0021 to 0.0076. As in the marine incident-free analysis, this range of impacts is the result of the analysis of two modes of spent nuclear fuel shipment, regularly scheduled commercial breakbulk vessels and chartered container vessels.

The highest estimate of incident-free maximally exposed worker risk is the same as for the basic implementation of Management Alternative 1 (0.00052 LCF).

Impacts of Accidents During Port Activities

Port accident risks were calculated based on the per-shipment risks determined in the analysis of the basic implementation of Management Alternative 1. The analysis examined the impact of using a wide range of ports based on the population around the port city, from high density population ports such as Elizabeth, NJ, to low-density ports such as the MOTSU terminal in North Carolina. The analysis also considered the impact of chartered shipments (no intermediate port stops before the vessel reaches the spent nuclear fuel port of entry) versus regularly scheduled commercial shipments with up to two intermediate ports of call before the spent nuclear fuel port of entry. Port accident risks associated with the Hybrid Alternative are estimated to range from 2×10^{-7} to 0.00002 LCF from radiation. The range of fatality estimates is due to both the differences in port city populations and the number of intermediate port stops.

of Management Alternative 1. The frequency is lower due to the reduced number of cask shipments, so the MEI risk is reduced to about 1 x 10^{-10} LCF.

4.5.3 Ground Transport Impacts

Impacts of Incident-Free Ground Transport

Radiological impacts of incident-free ground transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. The results are presented in Figure 4-20. The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.011 to 0.15 over the 13-year duration of the acceptance program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The range of fatality estimates is caused by two factors: the option of using truck or rail to transport spent nuclear fuel and the possibility of using different ports that created varying shipment distances.

Impacts of Accidents During Ground Transport

Transportation accident population risks over the entire Hybrid Alternative are estimated to range from 0.000005 to 0.000081 LCF from radiation and from 0.002 to 0.069 for traffic fatality, depending on the transportation mode and the ports that might be selected. The reason for the range of fatality estimates is the same as those described for incident-free transportation.

The maximum foreseeable offsite transportation accident is identical to that for the basic implementation of Management Alternative 1. The risk is reduced to 7.1×10^{-12} LCF due to the reduced amount of ground transport.

4.5.4 Management Site Impacts

Under the Hybrid Alternative, the amount of foreign research reactor spent nuclear fuel that would be accepted into the United States is about 17,100 elements and 13.9 MTHM. All the TRIGA spent nuclear fuel, representing approximately 4,900 elements and 1.0 MTHM, would be received and stored in existing facilities at the Idaho National Engineering Laboratory. Aluminum-based spent nuclear fuel, representing approximately 12,200 elements and 12.9 MTHM, would be received and chemically separated at the Savannah River Site as described in Implementation Alternative 6 to Management Alternative 1 (near term chemical separation in the United States). Environmental impacts associated with the receipt and storage of the TRIGA spent nuclear fuel at existing facilities at the Idaho National Engineering Laboratory would be covered by the impacts presented for the basic implementation of Management Alternative 1 without construction of new facilities (Section 4.2). Environmental impacts associated with the receipt and chemical separation of the aluminum-based spent nuclear fuel at the Savannah River Site would be covered by the impacts presented for the near-term chemical separation alternative at the Savannah River Site (Section 4.3.6). The occupational and public health and safety impacts for both sites were estimated by combining the appropriate results from earlier analyses for the Idaho National Engineering Laboratory and the Savannah River Site.

Impacts to the Public of Incident-Free Management Site Activities

The approximately 4,900 elements that would be received and managed at the Idaho National Engineering Laboratory under this alternative represent about 22 percent of the number of elements that would be received and managed there under the basic implementation of Management Alternative 1. Annual public impacts due to incident-free emissions from both aluminum-based and TRIGA foreign research reactor spent nuclear fuel during receipt and management at the Idaho National Engineering Laboratory under the basic implementation of Management Alternative 1 are presented in Table 4-9. Applying these results to the Hybrid Alternative at the Idaho National Engineering Laboratory for only TRIGA spent nuclear fuel is conservative because the TRIGA spent nuclear fuel would produce less gaseous fission product emissions than the mixture of spent nuclear fuel in the basic implementation of Management Alternative 1. Multiplying the results in Table 4-9 by the maximum duration of each activity (13 years for receipt and 40 years for storage) yields the highest estimated risks for this part of the Hybrid Alternative. The receipt/unloading impacts are reduced by the factor of 22 percent. The highest estimated public MEI risk is 7.8×10^{-10} LCF and the highest estimated public population risk is 0.0000064 LCF.

The approximately 12,200 elements that would be received at the Savannah River Site under this alternative represent about 54 percent of the number of elements that would be received and temporarily stored there under the basic implementation of Management Alternative 1. Annual public impacts due to incident-free emissions during receipt at the Savannah River Site under the basic implementation of Management Alternative 1 are presented in Table 4-8. The impacts for storage in RBOF are much smaller than those for receipt. Multiplying these results by 54 percent and the maximum duration of 13 years yields the highest estimated risks for this part of the Hybrid Alternative. The highest estimated public MEI risk is 3.9×10^{-10} LCF and the corresponding estimated public population risk is 0.000020 LCF.

The approximately 12.9 MTHM that would be chemically separated at the Savannah River Site under this alternative represents about 71 percent of the MTHM that would be chemically separated there under Implementation Alternative 6 dedicated to foreign research reactor spent nuclear fuel. Public impacts due to this implementation alternative were presented earlier in this chapter in Table 4-48. Multiplying these results by 71 percent yields the estimated impacts to the public near the Savannah River Site due to this part of the Hybrid Alternative. Using this procedure, the highest estimated incident-free public MEI risk at the Savannah River Site is 0.0000031 LCF. The highest estimated incident-free public population risk at the Savannah River Site (including both the air and water exposure pathways) is 0.13 LCF.

The maximum of the three onsite activities' estimated public incident-free MEI risks is equal to 0.0000031 LCF, which would result from chemical separation activities at the Savannah River Site (The three parts are receipt and management of TRIGA spent nuclear fuel at the Idaho National Engineering Laboratory, receipt and temporary management of aluminum-based spent nuclear fuel at the Savannah River Site, and chemical separation at the Savannah River Site). Thus, the chance of this individual incurring an LCF due to the Hybrid Alternative would be less than one in one hundred thousand.

The total of the three onsite activities' estimated public incident-free population risks is 0.13 LCF.

Impacts to Workers of Incident-Free Management Site Activities

Incident-free maximally exposed worker radiation dose depends upon the duration of the receipts, not the amount of spent nuclear fuel involved. The duration of this Hybrid Alternative is 13 years, the same as that in both the basic implementation of Management Alternative 1 and Implementation Alternative 6. Thus, the estimated maximally exposed worker dose is also the same. The maximally exposed worker risk is estimated to be 0.026 LCF.

Incident-free worker population impacts due to the basic implementation of Management Alternative 1 at the Idaho National Engineering Laboratory were presented in Section 4.2.4. Using the same evaluation process described in Appendix F, Section F.5, for the 162 casks of TRIGA foreign research reactor spent nuclear fuel that would be received and unloaded under this Hybrid Alternative yields a dose of 52 person-rem (dry storage in existing facilities). The associated worker population risk for this part of the Hybrid Alternative is 0.021 LCF.

Workers at the Savannah River Site would receive and unload 406 casks of aluminum-based foreign research reactor spent nuclear fuel in an existing wet facility under this alternative, receiving a population dose of 157 person-rem. The associated worker population risk for this part of the Hybrid Alternative is 0.063 LCF.

Incident-free worker population impacts due to Implementation Alternative 6 (chemical separation) were presented earlier in this chapter in Table 4-48. Multiplying these results by 71 percent yields the estimated incident-free impacts to the workers at the Savannah River Site due to the Hybrid Alternative. Using this procedure, the highest estimated incident-free worker population risk due to chemically separating this spent nuclear fuel at the Savannah River Site is 0.078 LCF.

The total of the three onsite activities' estimated incident-free worker population risks is 0.16 LCF.

Impacts of Accidents Onsite

Accident scenarios, frequencies, consequences, and annual risks for the Hybrid Alternative are derived from those for the basic implementation of Management Alternative 1 at the Idaho National Engineering Laboratory and Implementation Alternative 6 at the Savannah River Site.

Annual accident risks for receipt, unloading, and storage at the Idaho National Engineering Laboratory were presented earlier in this chapter in Table 4-25. Multiplying these by the duration of the activity (13 years for receipt and 40 years for storage) yields the

Table 4-61 Maximum Estimated Radiological Health Impacts of the Hybrid Alternative

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	Population	
		General Public	Workers
Marine Transport			
Incident-Free	0.00052	0	0.024
Accidents	3×10^{-10}	much less than 0.00002	---
Port Activities			
Incident-Free	0.00052	0	0.0076
Accidents	1×10^{-10}	0.00002	---
Ground Transport			
Incident-Free	0.00052	0.11	0.037
Accidents	7.1×10^{-12}	0.000081	---
Site Activities			
Incident-Free	0.026	0.13	0.16
Accidents	0.000033	0.34	---
Highest Individual Risk			
Incident-Free	0.026	----	----
Accidents	0.000033	----	----
Total Population Risk			
Incident-Free	----	0.24	0.23
Accidents	----	0.34	----

Table 4-61 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE management site receiving the maximum dose allowed by DOE regulation every year.

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 7.8×10^{-8} LCF.

The highest estimated accident MEI risk is 0.000033 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than one in ten thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

As shown in Table 4-61, the total incident-free population risk would be 0.24 LCF for the potentially exposed public, while the corresponding risk would be 0.23 LCF for workers. Thus, there would be an estimated 24 percent chance of incurring one additional LCF among the exposed general public, and a 23 percent chance of incurring one additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-61. There is about a seven percent chance that a truck driver or member of the public could die in a traffic accident associated with this Hybrid Alternative. This death would be unrelated to the radioactive nature of the cargo.

4.6 No Action Alternative

Under the No Action Alternative, no foreign research reactor spent nuclear fuel or high-level waste would be accepted into or managed by the United States. The United States would not provide any technical or financial assistance to foreign research reactor operators for the management of their spent nuclear fuel. The United States would rely on the foreign governments' compliance with existing international agreements to control the disposition of foreign research reactor spent nuclear fuel containing uranium enriched in the United States.

Policy Considerations

The No Action Alternative would have a major adverse impact on U.S. nuclear weapons nonproliferation policy. The No Action Alternative would not remove any of the approximately 4.6 metric tons of U.S. origin HEU from international commerce as considered under the proposed action. Under this alternative, the foreign research reactor owners would continue, or may revert back to, use of HEU fuel in their reactors. Countries that can reprocess might send their HEU spent nuclear fuel to be reprocessed and use the separated HEU to produce fresh HEU fuel. In addition, any new research reactors to be built would likely be designed to use HEU fuel. Thus, the No Action Alternative could cause an increase in the number of shipments of weapons-grade nuclear material in transit around the world. It would also damage, perhaps irreparably, the credibility of the RERTR program. Countries that cannot reprocess their research reactor spent nuclear fuel would have to store their fuel. As the spent nuclear fuel ages, it becomes less dangerous to handle (its radioactivity decreases with time), and could possibly become a target of theft and diversion. Hence, the No Action Alternative would undermine the U.S. nuclear weapons nonproliferation policy and the risk of weapons-grade nuclear material being diverted into a nuclear weapons program would increase markedly.

To demonstrate the risk of having reactor owners continue, or revert back to, use of HEU fuel, please see Tables B-3, B-4, and B-5 in Appendix B. These tables list the 104 foreign research reactors whose spent nuclear fuel is included under the proposed action, including 24 reactors that have been converted (fully or partially) or are in various stages of conversion (i.e., ordered, or anticipated to begin converting) from HEU to LEU fuel, and 30 reactors that could be converted, but are not being converted, because the owners of the research reactors are awaiting the outcome of this EIS before they make a decision. Under the No Action Alternative, it is possible that up to 48 foreign research reactor operators could choose to continue or revert back to using HEU fuel in their reactors. These tables also list 23 foreign research reactor operators who possess HEU spent nuclear fuel, even though their reactors are either already shut down or planned to be shutdown for various reasons. This HEU spent nuclear fuel would remain in the foreign research reactor host countries, if the No Action Alternative is selected.

On the other side of the ledger, the benefits obtained from research reactors, described briefly in Section 1.1 of the EIS, would be diminished. Since the No Action Alternative means no U.S. assistance to foreign research reactor operators for managing their spent nuclear fuel, additional research reactors may be forced to shut down, because of lack of funds and/or long term storage capabilities. DOE and the Department of State cannot estimate the number of reactors that would actually be shut down because this would depend on each country's regulations regarding spent nuclear fuel storage. Nevertheless, the medical, industrial and environmental services provided by the shutdown research reactors would be lost. For medical services in particular, foreign research reactors produce radioisotopes used in nuclear medicine in the United States (as discussed in Sections 1.1 and 4.3.1.3 of the EIS). If some of these reactors were forced to shut down, a shortage of medical radioisotopes could occur in the United States. Since the U.S. medical requirements for radioisotopes are not likely to decrease in the near future, alternative sources would have to be found. This could involve an increased level of activity at existing U.S. research reactors or construction of a new reactor in the United States to supply the needed medical radioisotopes, with all the potential environmental impacts of these actions.

Environmental Impacts of Overseas Storage without U.S. Assistance

The policy considerations, marine transport impacts, port activities impacts, ground transport impacts, and management site impacts of the preferred alternative presented in this section are based on analysis performed for the basic implementation of Management Alternative 1 (Section 4.2), Implementation Alternative 1c (Section 4.3.1.3), Implementation Alternative 6 (Section 4.3.6), and Implementation Alternative 7 (Section 4.3.7).

4.7.1 Policy Considerations

The policy considerations for the preferred alternative are similar to those described in Section 4.2 for Management Alternative 1. A critical result of implementing this preferred alternative would be the

This potential method of handling the foreign research reactor spent nuclear fuel would be consistent with United States nonproliferation policy, despite the use of chemical separation, because (1) it would reduce the worldwide stockpiles of this nuclear weapons material; (2) no plutonium would be separated; and (3) the chemical separation would not be taking place for either nuclear weapons or nuclear power purposes.

DOE is aware that the inclusion of chemical separation within the preferred alternative could be interpreted by some nations, organizations, and persons as a signal of endorsement of the use of chemical separation as a routine method of waste management for spent nuclear fuel or a reversal of United States policy on chemical separation. This would not be an accurate interpretation. The United States policy regarding chemical separation was established in Presidential Decision Directive 13, and DOE and the Department of State have determined that this preferred alternative is consistent with that policy. The draft version of this EIS indicated that chemical separation is a non-preferred technology. This final preferred alternative includes provision for possible chemical separation. DOE maintains a presumption that spent nuclear fuel would not be chemically separated unless there is an imminent health and safety risk, or other programmatic conditions, that cannot be addressed during the time period when no feasible alternative to chemical separation is available. These considerations will be addressed by the independent study described in Section 2.9.

4.7.2 Marine Transport Impacts

The marine transport impacts of the preferred alternative would be similar to those of the basic implementation of Management Alternative 1, with the addition of the target material shipments. As discussed in Section 4.3.1.3 and Appendix B, Section B.1.5, target material would be prepared for transport by changing it into either oxide or calcine form, and both forms might be accepted at some time during the proposed policy period. Even though it requires less marine transport, the oxide form presents a higher radiological risk under accident conditions because its smaller particle size is more easily dispersed in air. Therefore, to be conservative, the analysis of marine and port radiological accidents is based on the assumption that all the target material would be shipped as an oxide. The rest of the marine and port target material transport analysis is based on the assumption of 15 cask shipments, which is the maximum number of marine target material casks. This represents an increase of approximately two percent over the 721 marine cask shipments in the basic implementation of Management Alternative 1.

Marine transport to the West Coast of the United States would be limited to a maximum of approximately 38 casks, which slightly decreases the total number of days the ships would be at sea. Furthermore, DOE would strive to minimize the number of shipments necessary by coordinating shipments from several reactors at a time (i.e., by placing multiple casks [up to 8] on a ship). DOE currently estimates that approximately 5 shipments through the Naval Weapons Station at Concord, California would be necessary.

Impacts of Incident-Free Marine Transport

The highest estimated maximally exposed worker risk due to foreign research reactor spent nuclear fuel is 0.00052 LCF, which is based on the conservative assumption that one individual receives the maximum annual dose (100 mrem) every year for 13 years (Table 4-2). This means that the chance of this

Target material contains far less radioactivity than foreign research reactor spent nuclear fuel. Each transportation cask of target material would produce a radiation dose rate far below the rate that was assumed for the foreign research reactor spent nuclear fuel. Thus, the rounded-off results of the incident-free radiological risk calculations for the basic implementation of Management Alternative 1 are not affected by the addition of up to 15 marine casks of target material.

Impacts of Accidents During Marine Transport

The risks associated with accidents at sea are bounded by the risks of the same accidents in ports because humans in the vicinity of accidents at sea are much fewer in number than even the least populated port.

Marine Transport Cumulative Impacts and Mitigation Measures

The marine transport cumulative impacts and mitigation measures for the preferred alternative would be the same as for the basic implementation of Management Alternative 1, which are discussed in Sections 4.2.1.4 and 4.2.1.5, respectively.

4.7.3 Port Activities Impacts

Although all of the candidate ports of entry presented in Section 3 are acceptable, based on the port selection criteria described in Appendix D, DOE would prefer to use military ports. All aluminum-based foreign research reactor spent nuclear fuel and target material from overseas would arrive at candidate ports on the East Coast of the United States, preferably the Naval Weapons Station at Charleston, South Carolina. Up to approximately 38 casks of TRIGA foreign research reactor spent nuclear fuel would arrive at candidate ports on the West Coast of the United States, preferably the Naval Weapons Station at Concord, California.

Impacts of Incident-Free Port Activities

As shown in Table 4-5, the highest maximally exposed worker risk is 0.00052 LCF, which is based on the conservative assumption that one individual receives the maximum annual dose (100 mrem) every year for 13 years. This means that the chance of this hypothetical individual incurring a latent cancer due to the preferred alternative would be less than one in a thousand.

The highest estimated population risk for port workers is about 0.012 LCF, as discussed in Section 4.2.2.3.

As discussed under *Impacts of Incident-Free Marine Transport* above, each transportation cask of target material would produce a radiation dose rate far below the rate that was assumed for the foreign research reactor spent nuclear fuel. Thus, the rounded-off results of the incident-free radiological risk calculations for the basic implementation of Management Alternative 1 are not affected by the addition of up to 15 cask shipments of target material.

Impacts of Accidents During Port Activities

The radiological risks due to port accidents were estimated in the same manner as for the basic implementation (Section 4.2.2.3) and Implementation Alternative 1c (Section 4.3.1.3) of Management Alternative 1. The highest estimated population risk for the entire preferred alternative program is 7.1×10^{-7} LCF. This risk estimate is lower than the earlier alternatives due to the use of military ports in the preferred alternative. These ports are located in areas of low population density, so the number of people potentially affected is much lower. The addition of target material causes a very small incremental increase (3×10^{-9} LCF) in the risk.

Port Activities Cumulative Impacts, Mitigation Measures, and Environmental Justice

The port activities cumulative impacts, mitigation measures, and environmental justice for the preferred alternative would be the same as for the basic implementation of Management Alternative 1, which are discussed in Sections 4.2.2.4, 4.2.2.5, and 4.2.2.6, respectively.

4.7.4 Ground Transport Impacts

The ground transport impacts were calculated under the assumption that only military ports would be used. DOE has selected military ports close to the management sites (the Charleston NWS in South Carolina and the Concord NWS in California) as the preferred ports of entry.

The risk estimates were maximized by assuming all target material would be oxide for radiological accident calculations and calcine for all other calculations. The calcine form could require up to 125 casks of target material to be shipped overland from Canada.

The preferred points of entry, destinations, and approximate numbers of cask shipments in the preferred alternative are presented in Table 4-62. Other shipment distributions would also be possible.

Table 4-62 Points of Entry, Destinations, and Numbers of Shipments in the Preferred Alternative

Cargo and Destination	*Point of Entry*			*Total Cask Shipments*
	East Coast	*West Coast*	*Canadian Border*	
Aluminum-Based Foreign Research Reactor Spent Nuclear Fuel to the Savannah River Site	559	0	116	675
TRIGA Foreign Research Reactor Spent Nuclear Fuel to the Idaho National Engineering Laboratory	124	38	0	162
Target Material to the Savannah River Site	up to 15	0	up to 125	up to 140
Total Cask Shipments	**up to 698**	**38**	**up to 241**	**up to 977**

Impacts of Incident-Free Ground Transport

The incident-free ground transport of foreign research reactor spent nuclear fuel and target material is estimated to result in a maximum of 0.089 LCF over the entire duration of the program. This is the sum of the estimated number of radiation-related LCF to the public and transportation workers.

The estimated maximum number of radiation-related LCF for transportation workers is 0.022. The estimated maximum number of radiation-related LCF for the general public is 0.067, and the estimated maximum number of non-radiation-related fatalities from vehicular emissions is 0.018.

Impacts of Accidents During Ground Transport

The total ground transport accident population risks for the preferred alternative are estimated to be less than 0.00072 LCF from radiation and 0.052 from traffic collisions.

The maximum foreseeable offsite transportation accident would involve a transportation cask of oxide target material in a suburban population zone under neutral (average) weather conditions, which could expose the MEI to 150 mrem. A similar event involving a transportation cask of spent nuclear fuel could expose the MEI to 2.4 mrem. These events are both in the highest accident severity category. Taking all the possible consequences and frequencies of these accidents into account, and adding the foreign research reactor spent nuclear fuel risks with the target material risks yields the MEI risk of 2.7 x 10^{-11} LCF for the preferred alternative.

Ground Transport Cumulative Impacts, Mitigation Measures, and Environmental Justice

The ground transport cumulative impacts, mitigation measures, and environmental justice for the preferred alternative would be the same as for the basic implementation of Management Alternative 1, which are discussed in Sections 4.2.3.4, 4.2.3.5, and 4.2.3.7, respectively.

4.7.5 Management Site Impacts

As discussed in Section 2.9, all the TRIGA foreign research reactor spent nuclear fuel would be managed at the Idaho National Engineering Laboratory. The fuel would be received and stored in existing facilities. The environmental impacts of the preferred alternative at the Idaho National Engineering Laboratory can be estimated from the environmental impact analysis presented for the basic implementation of Management Alternative 1 (Section 4.2) socioeconomics, cultural resources, aesthetic and scenic resources, geology, air and water quality, ecology, noise, materials and energy consumption, and non-radiological or non-toxic waste production during construction.

The occupational and public health and safety, waste management, and cumulative impacts presented below assume that the implementation of the preferred alternative at the Savannah River Site would result in radiological health effects equal to those presented in Sections 4.3.6 and 4.3.7 of this EIS.

4.7.5.1 Occupational and Public Health and Safety

Impacts to the Public of Incident-Free Management Site Activities

The approximately 4,900 foreign research reactor spent nuclear fuel elements that would be received and managed at the Idaho National Engineering Laboratory under the preferred alternative represent about 22 percent of the total number of elements that would be received and managed there under the basic implementation of Management Alternative 1. Annual public impacts due to incident-free emissions from both aluminum-based and TRIGA foreign research reactor spent nuclear fuel during receipt and management at the Idaho National Engineering Laboratory under the basic implementation of Management Alternative 1 are presented in Table 4-9. Applying these results to the preferred alternative at the Idaho National Engineering Laboratory for only TRIGA spent nuclear fuel is conservative because the TRIGA spent nuclear fuel would produce less gaseous fission product emissions than the mixture of [illegible]

Impacts to Workers of Incident-Free Management Site Activities

Incident-free maximally exposed worker radiation dose depends upon the duration of the receipts, not the amount of spent nuclear fuel involved. The duration of the receipts in the preferred alternative is 13 years, the same as that in the basic implementation, the target material alternative, and the chemical separation alternative of Management Alternative 1. Thus, the estimated maximally exposed worker dose is also the same. The highest maximally exposed worker risk is estimated to be 0.026 LCF.

The incident-free worker population risks of the basic implementation of Management Alternative 1 at the Idaho National Engineering Laboratory were presented in Section 4.2.4.1. Using the same evaluation process yields a dose of 52 person-rem (dry storage in existing facilities). The associated worker population risk for this part of the preferred alternative is 0.021 LCF.

The incident-free radiological worker health impacts at the Savannah River Site due to the preferred alternative are assumed to be equal to those discussed in Section 4.3.6.6.4 under the subheading, *Incident-Free Impacts at the Savannah River Site*. The highest estimated worker population risk is 0.21 LCF.

The total of the onsite activities' estimated incident-free worker population risks at both sites is 0.23 LCF, which means that there would be an approximately 23 percent chance of one additional LCF among the affected radiation workers at the two sites.

Impacts to the Public of Accidents Onsite

Accident scenarios, frequencies, consequences, and risks for the preferred alternative at the Idaho National Engineering Laboratory are the same as those for the basic implementation of Management Alternative 1.

4.7.5.2 Waste Management

Implementation of the receipt and storage portions of the preferred alternative would introduce a very small increase in waste generation over current levels at both sites. Baseline site generation of waste is shown in Appendix F, Tables F-23 and F-46 for the Savannah River Site and the Idaho National Engineering Laboratory, respectively. It should be noted that the figures represent storage of more fuel elements, at both sites, than the amounts indicated by the preferred alternative. Implementation of a new technology would produce waste in the amounts presented in Table 4-56.

If the chemical separation portion of the preferred alternative is implemented, this would generate different wastes at the Savannah River Site in place of some of the waste from the new technology. As discussed in Section 4.3.6.6.5, the primary wastes generated during conventional chemical separation and vitrification operations are high-level waste glass in canisters and saltstone. Assuming the chemical separation portion of the preferred alternative could involve up to approximately one-third of the aluminum-based foreign research reactor spent nuclear fuel (6,000 elements), this waste generation would be about one-third of the amount generated under Implementation Alternative 6. Under the preferred alternative, DOE could generate up to approximately 24 high-level waste glass canisters and 1,350 cubic meters (47,700 cubic feet) of saltstone. These wastes would be managed along with much larger quantities of identical wastes in existing facilities at the Savannah River Site.

4.7.5.3 Cumulative Impacts

Cumulative impacts from the implementation of the preferred alternative at both the Idaho National Engineering Laboratory and the Savannah River Site are expected to be lower than those presented for the basic implementation of Management Alternative 1 in Sections 4.2.4.3.1 and 4.2.4.3.2 for the two sites, respectively. At both sites the cumulative impacts from the management of foreign research reactor spent nuclear fuel and impacts from other existing or planned activities or actions at the sites, as presented in Tables 4-29 and 4-30 for Savannah River Site and Idaho National Engineering Laboratory, respectively, including activities not related to the management of spent nuclear fuel, would not challenge or have detrimental effects on the public or environmental resources at the sites.

4.7.5.4 Mitigation Measures

Although environmental impacts at both the Savannah River Site and the Idaho National Engineering Laboratory for the implementation of the preferred alternative would be minimal in all environmental media and mitigation measures would not be necessary, the sites would implement measures in some areas to minimize impacts. Such measures would be taken in the areas of pollution control, socioeconomics, cultural resources, air and water resources, occupational and public health and safety, and accident prevention. Section 4.2.4.6 provides details on these issues.

4.7.5.5 Environmental Justice

The environmental justice conclusions for the management sites discussed in Section 4.2.4.5 for the implementation of Management Alternative 1 are valid for the preferred alternative. As discussed in Section 4.2.4.5, minority or low-income populations living near the Savannah River Site or the Idaho National Engineering Laboratory would not be subjected to any disproportionately high and adverse impacts.

4.7.6 Short Term Uses and Long Term Productivity

The use of land at the Savannah River Site for the potential construction of the new technology facilities would conform with the land use policy at the site. After adoption of an overall strategy for the management of all DOE-owned spent nuclear fuel (including spent nuclear fuel from foreign research reactors), some of the areas may be released for other productive uses.

4.7.7 Irreversible and Irretrievable Commitments of Resources

The operation of existing storage facilities at both sites would involve the consumption of some irretrievable amounts of electrical energy. The potential construction of new technology facilities at the Savannah River Site would consume irretrievable amounts of electrical energy, fuel, concrete, sand, and gravel. Other resources used in the construction would probably not be recoverable. These would include finished steel, aluminum, copper, plastics, and lumber. Most of this material would be incorporated in foundations, structures, and machinery.

4.7.8 Summary of the Impacts of the Preferred Alternative

The principal impacts of the preferred alternative would be occupational and public health and safety impacts. These are presented in Table 4-63 in terms of the risk of death due to cancer during each of the four segments of this alternative. The table also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF due to the preferred alternative. The population risk expresses the estimated number of additional LCF among the entire potentially exposed population.

Table 4-63 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the management site receiving the maximum dose allowed by DOE regulation every year.

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free risk for individual members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 0.0000043 LCF.

The highest estimated accident MEI risk is 0.000047 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to an accident under this alternative would be less than one in ten thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

As shown in Table 4-63, the total incident-free population risk would be 0.25 LCF for the potentially exposed public, while the corresponding risk would be 0.30 LCF for workers. Thus, there would be an estimated 25 percent chance of incurring one additional LCF among the exposed general public, and a 30 percent chance of incurring one additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Table 4-63 Maximum Estimated Radiological Health Impacts of the Preferred Alternative

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	Population	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.00052	0	0.034
Accidents	5×10^{-10}	much less than 7.1×10^{-7}	---
Port Activities			
Incident-Free	0.00052	0	0.012
Accidents	2.9×10^{-10}	7.1×10^{-7}	---
Ground Transport			
Incident-Free	0.00052	0.067	0.022
Accidents	2.7×10^{-11}	0.00072	---
Site Activities			
Incident-Free	0.026	0.18	0.23
Accidents	0.000047	0.45	---
Highest Individual Risk			
Incident-Free	0.026	----	---
Accidents	0.000047	----	----
Total Population Risk			
Incident-Free	----	0.25	0.30
Accidents	----	0.45	----

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-63. There is approximately a five percent chance that a truck driver or member of the public could die in a traffic accident associated with the preferred alternative. This death would be unrelated to the radioactive nature of the cargo.

4.8 Comparison of the Alternatives

This chapter has identified the policy considerations and potential environmental impacts resulting from the proposed action, with all of its various alternatives, and the No Action Alternative. This section provides a comparison of the potential impacts of each alternative, with emphasis on key issues such as the amount of HEU removed from international commerce and risks to workers and the public.

4.8.1 Amount of HEU Removed from International Commerce

The purpose and need for Agency action is driven by the concern that HEU in civilian commerce might be diverted into a nuclear weapons program. Removal of HEU from international civilian commerce will greatly enhance the goals of the U.S. nuclear weapons nonproliferation policy. Figure 4-21 compares the quantities of HEU that would be removed from international civil commerce under the basic implementation of Management Alternative 1, the implementation alternatives, the Hybrid Alternative, the No Action Alternative, and the preferred alternative.

Basic Implementation of Management Alternative 1: The basic implementation of Management Alternative 1 would remove up to an estimated 4.6 metric tons (5.1 tons) of HEU from international commerce. By accepting this weapons-grade material into the United States for storage, the risk of material diversion would be eliminated. For comparison, the United States moved about 0.6 metric tons (0.7 tons) of HEU from Kazakhstan to the United States in November and December 1994 to ensure that it could not be diverted into a nuclear weapons program. The quantity of HEU involved in the basic implementation of Management Alternative 1 is over seven times the amount removed from Kazakhstan. The HEU in foreign research reactor spent nuclear fuel, however, is mixed with fission products, so it would require more sophisticated chemical processing to convert it to uranium metal suitable for use in nuclear weapons.

Implementation Alternatives: Acceptance of amounts of foreign research reactor spent nuclear fuel different from the amounts identified in the basic implementation of Management Alternative 1 could have an impact on the amount of HEU in international civil commerce. As shown in Figure 4-21, the implementation alternative of accepting spent nuclear fuel only from developing nations would remove up to only about 0.24 metric tons (0.26 tons) of HEU from international commerce. The implementation alternative of accepting target material in addition to the foreign research reactor spent nuclear fuel in the basic implementation of Management Alternative 1 would remove the most HEU (up to 4.8 metric tons or 5.3 tons) from international commerce. If the acceptance policy lasted for only 5 years, then the amount of HEU involved would be only up to 4.1 metric tons (4.5 tons).

Implementation through financial arrangements different from those identified in the basic implementation of Management Alternative 1 could indirectly impact the amount of HEU removed from international commerce depending on whether those financial adjustments influence the amount of foreign research

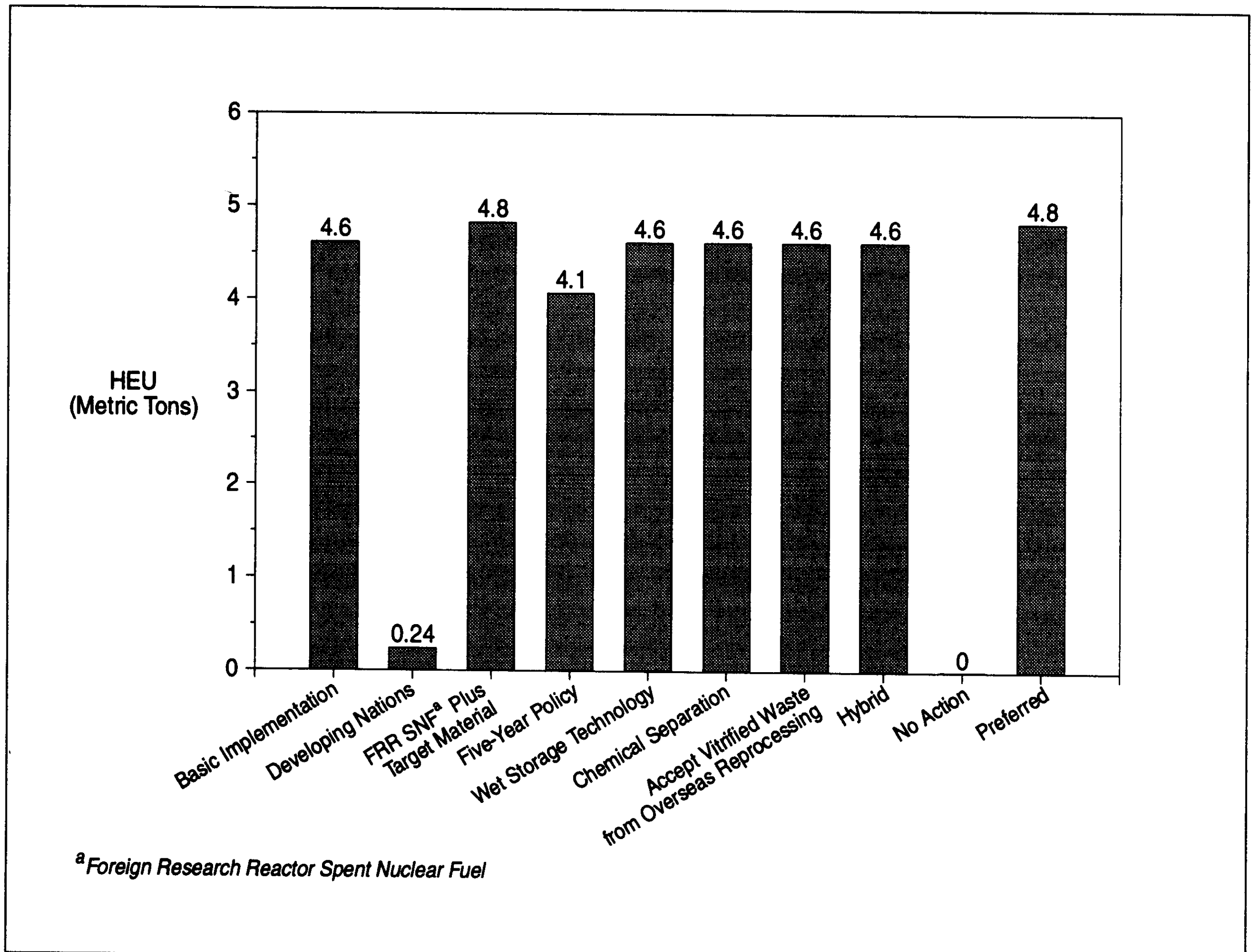


Figure 4-21 Quantities of HEU that Would Be Removed from International Commerce Under Each Alternative

reactor spent nuclear fuel transported to the United States. The final amount of HEU removed from international civil commerce through the application of different financial arrangements cannot be readily determined at this point.

Implementation by taking title to the foreign research reactor spent nuclear fuel at locations different from those identified in the basic implementation of Management Alternative 1 would not change the amount of HEU removed from international commerce, i.e., the action would still remove up to 4.6 metric tons (5.1 tons) of HEU. Similarly, the use of wet storage technology for the interim period instead of dry storage technology as identified in the basic implementation of Management Alternative 1 would not change the amount of HEU removed from international civil commerce, since the alternative relates to actions within the United States. Implementation by use of near term chemical separation in the United States instead of interim storage would also cause no change in the amount of HEU removed, again because the alternative involves actions in the United States.

Storing foreign research reactor spent nuclear fuel at one or more overseas sites would have a questionable effect on the amount of HEU removed from international commerce. Although this management alternative would provide the United States some limited measure of control over the foreign research reactor spent nuclear fuel, the prevention of material diversion into a nuclear weapons program would not be as fully ensured as if the foreign research reactor spent nuclear fuel was accepted into the United States. This alternative would leave HEU stockpiled around the world.

The implementation alternative of overseas reprocessing would remove the same amount of HEU from international commerce as would the basic implementation of Management Alternative 1, independent of decisions on the management of the resulting high-level waste.

Hybrid Alternative: The Hybrid Alternative chosen for analysis would remove the same amount of HEU from international commerce as would the basic implementation of Management Alternative 1, independent of decisions on the management of the resulting high-level waste.

No Action Alternative: Under this alternative, the United States would rely solely on the foreign governments' compliance with international agreements to control the foreign research reactor spent nuclear fuel. A policy of no action by DOE and the Department of State runs counter to U.S. nuclear weapons nonproliferation policy by causing continued reliance on HEU, thus not realizing the goal of eliminating civil commerce in HEU.

Preferred Alternative: The preferred alternative would remove the same amount of HEU (up to 4.8 metric tons or 5.3 tons) from international commerce as would Implementation Alternative 1c of Management Alternative 1. This amount is higher than for the other alternatives.

4.8.2 Radiological Risk to Individuals

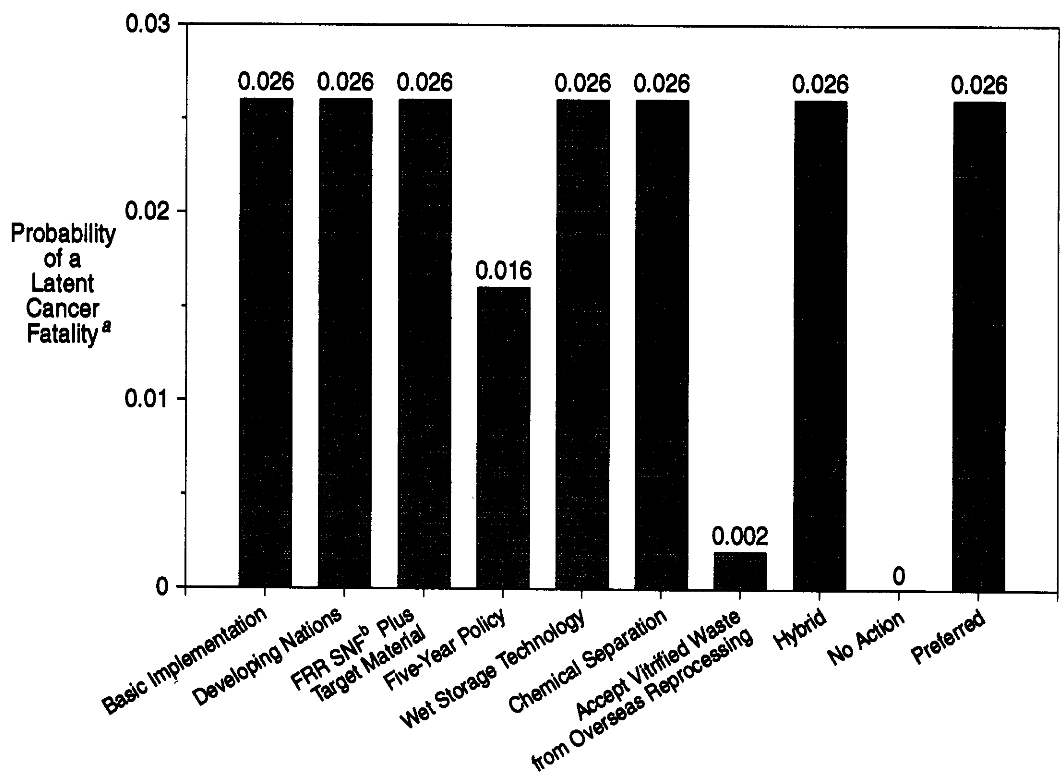
A maximally exposed worker or an MEI in the public is a hypothetical individual who records the highest possible exposure to radiation in a given situation, and the associated risks are different depending on the alternative considered. Figures 4-22 and 4-23 present comparisons of the estimated radiological risk to the maximally exposed worker and to the MEI under each alternative for incident-free and accident conditions, respectively. Alternatives involving the smallest number of cask shipments into the United States would produce the lowest individual risks. There would be no maximally exposed worker risk or MEI risk in the United States under the No Action Alternative.

The incident-free maximally exposed worker risk estimates are driven by the assumption that a radiation worker would receive the maximum radiation dose allowed by law for every year that foreign research reactor spent nuclear fuel is accepted. This risk depends only on the duration of the action, not on the number of casks or elements. Thus, the Five-Year Acceptance Alternative would present lower risk than the alternatives which last for 13 years.

The accident MEI risk estimates are dominated by onsite accident scenarios. This is because during marine transport, port activities, and ground transport, the foreign research reactor spent nuclear fuel would be inside transportation casks. During onsite activities, while spent nuclear fuel is outside of transportation casks, the probability of an incident that could release radioactive material is higher. The highest estimated accident MEI risk in the public is 0.00015 LCF, which means that this hypothetical individual's increased chance of incurring an LCF would be less than two in ten thousand.

4.8.3 Radiological Risk to Exposed Populations

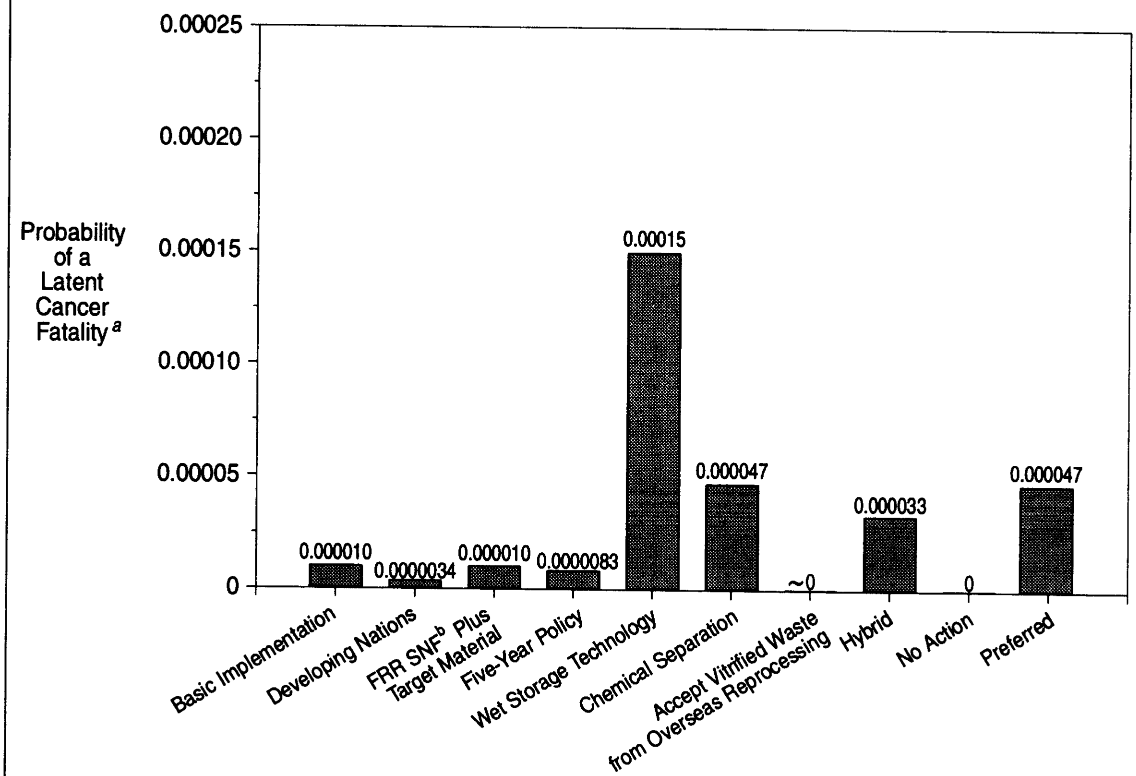
Population risk is the risk of additional latent cancers occurring among people (both public and workers) who would be exposed to radiation. Risks vary with the alternative considered. Figures 4-24 and 4-25 present comparisons of the estimated incident-free radiological risks to the public and worker populations under each alternative. Alternatives involving the smallest number of cask shipments into the United



[a] *Impacts evaluated were those in the United States and on the Global Commons.*

[b] *Foreign Research Reactor Spent Nuclear Fuel*

Figure 4-22 Maximum Estimated Incident-Free Radiological Risk to the Maximally Exposed Worker Under Each Alternative



[a] *Impacts evaluated were those in the United States and on the Global Commons.*

[b] *Foreign Research Reactor Spent Nuclear Fuel*

Figure 4-23 Maximum Estimated Accident Radiological Risk to the MEI in the Public Under Each Alternative

States would produce the lowest population risks. The chemical separation, overseas reprocessing, and preferred alternative are the alternatives in which the waste would be conditioned for disposal. Under the other alternatives, some form of processing may be required at some time in the future before disposal. There would be no population risk in the United States under the No Action Alternative. Under all the alternatives the estimated incident-free public and worker population risks would result in less than one-half additional LCF among each population group.

Figure 4-26 presents a comparison of the estimated accident radiological population risks to the public under each alternative. Those alternatives involving some form of processing in the United States would present the largest accident risks, but these risks would occur in the near term. Under the other alternatives, some form of processing may be required at some time in the future before disposal. Under all the alternatives, the estimated accident public population risks would result in less than one-half additional LCF.

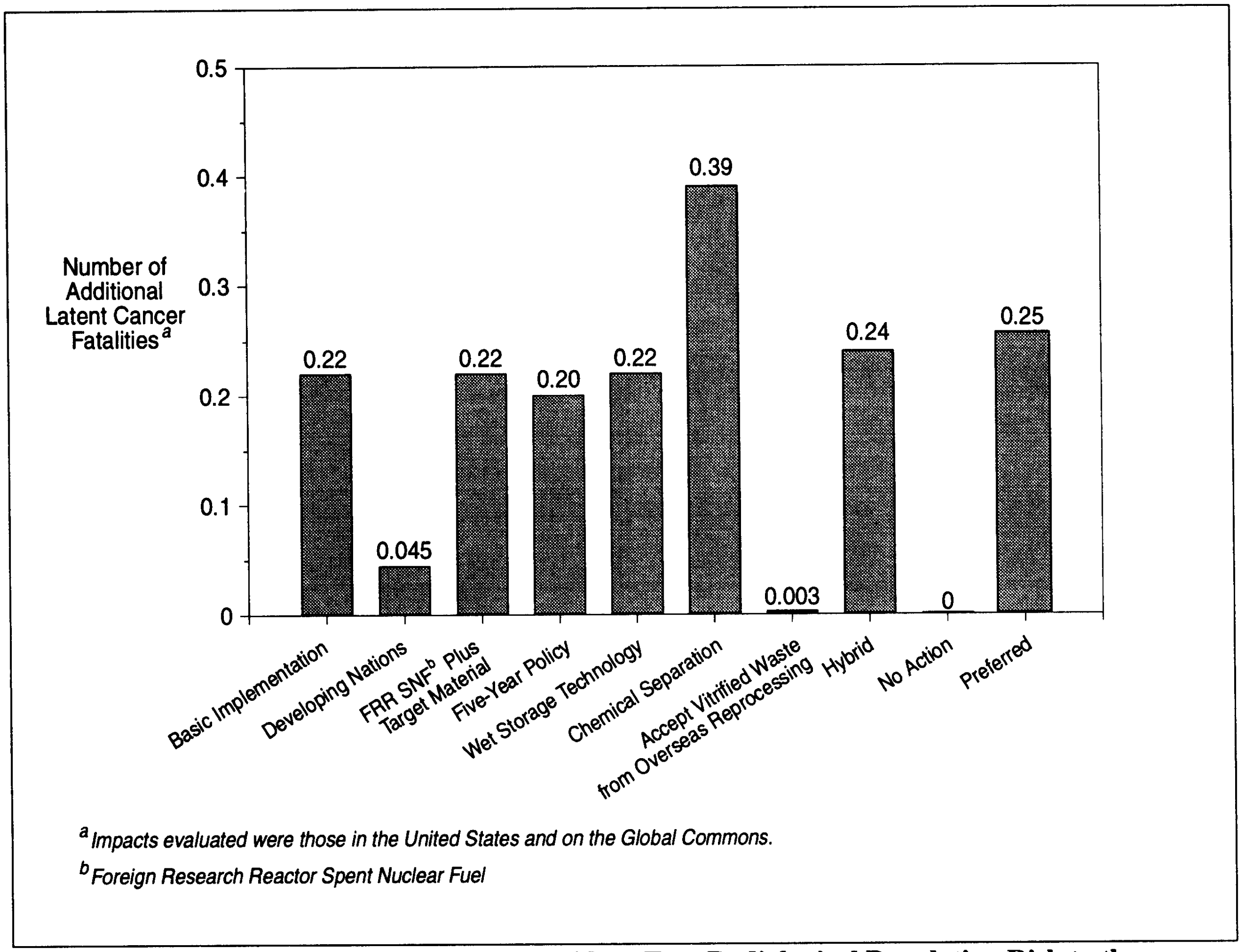


Figure 4-24 Maximum Estimated Incident-Free Radiological Population Risk to the General Public Under Each Alternative

4.8.4 Nonradiological Risks

The transport of foreign research reactor spent nuclear fuel from the ports to the sites would involve some risk of death due to traffic accidents for both the truck drivers and the public. Figure 4-27 presents a comparison of the estimated traffic accident risk to both the drivers and public combined under each alternative. Estimates include the risks associated with transporting the empty casks back to the ports.

Results are directly proportional to the number of highway miles over which casks would be transported under each alternative. The basic implementation of Management Alternative 1 and four of the implementation alternatives would have essentially the same risk, while the Developing Nations Subalternative and the Hybrid Alternative would have lower traffic accident risks.

Under the subalternative of accepting vitrified waste from overseas reprocessing, an estimated eight cask shipments would be accepted in the United States, so the traffic accident risk would be extremely low. There would be no population risk in the United States under the other overseas subalternative, as well as the No Action Alternative.

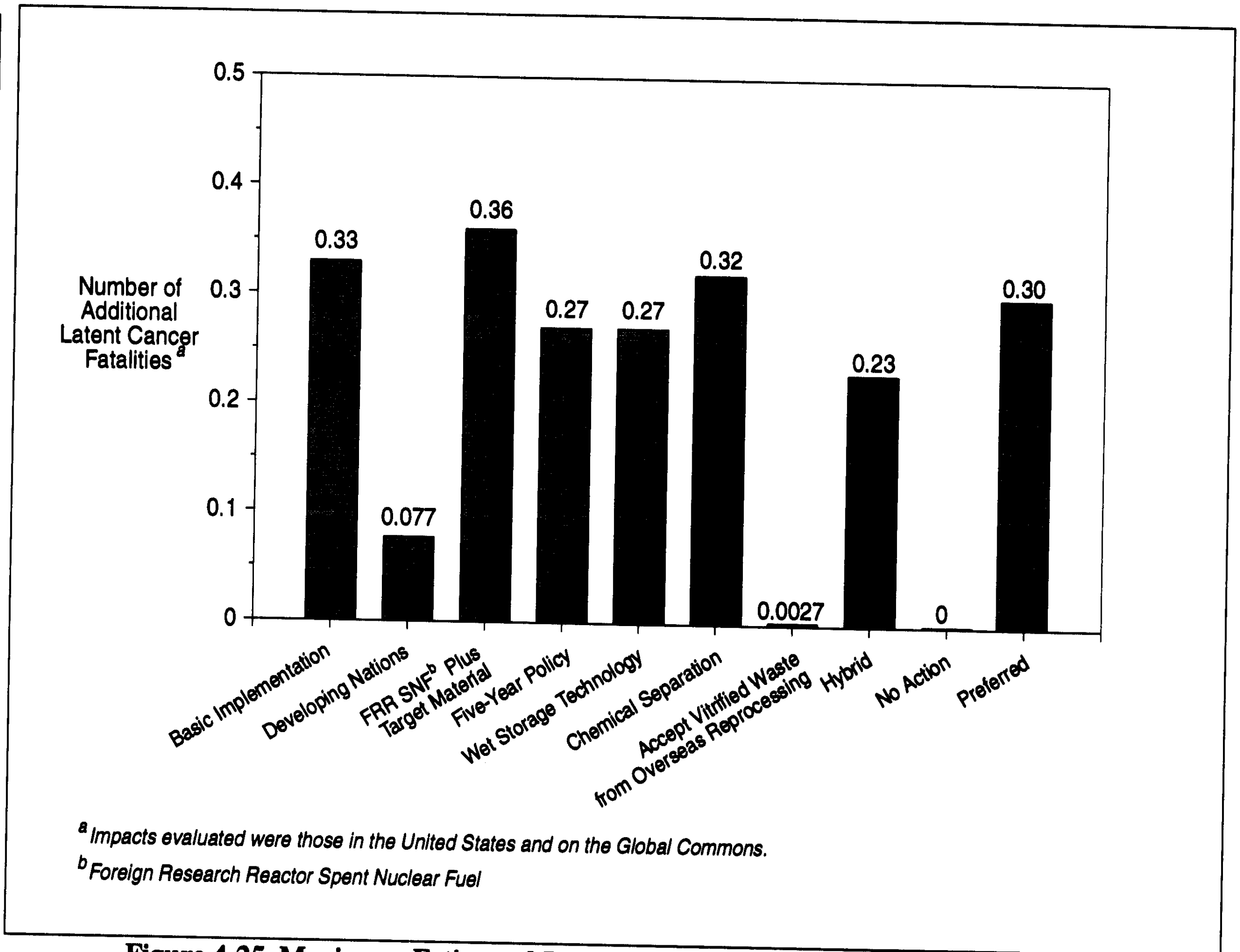


Figure 4-25 Maximum Estimated Incident-Free Radiological Population Risk to Workers Under Each Alternative

The traffic accident risk is also relatively low under the preferred alternative because all the cask shipments of aluminum-based foreign research reactor spent nuclear fuel would go through an east coast port or ports to the Savannah River Site. This effectively minimizes the ground transport risk by minimizing the number of highway miles required.

4.8.5 Land Use

Basic Implementation of Management Alternative 1: The basic implementation of Management Alternative 1 would not result in major land use issues at any of the potential foreign research reactor spent nuclear fuel management sites. If additional storage space were required for the foreign research reactor spent nuclear fuel, the space would be built on DOE-owned lands, inside the boundaries of the DOE management sites.

Implementation Alternatives: Acceptance of amounts of foreign research reactor spent nuclear fuel different from the amount identified in the basic implementation of Management Alternative 1 would not cause land use issues, even though storage needs may vary due to the United States receiving a larger (if target material is accepted in addition to spent nuclear fuel) or smaller (e.g., from developing nations only)

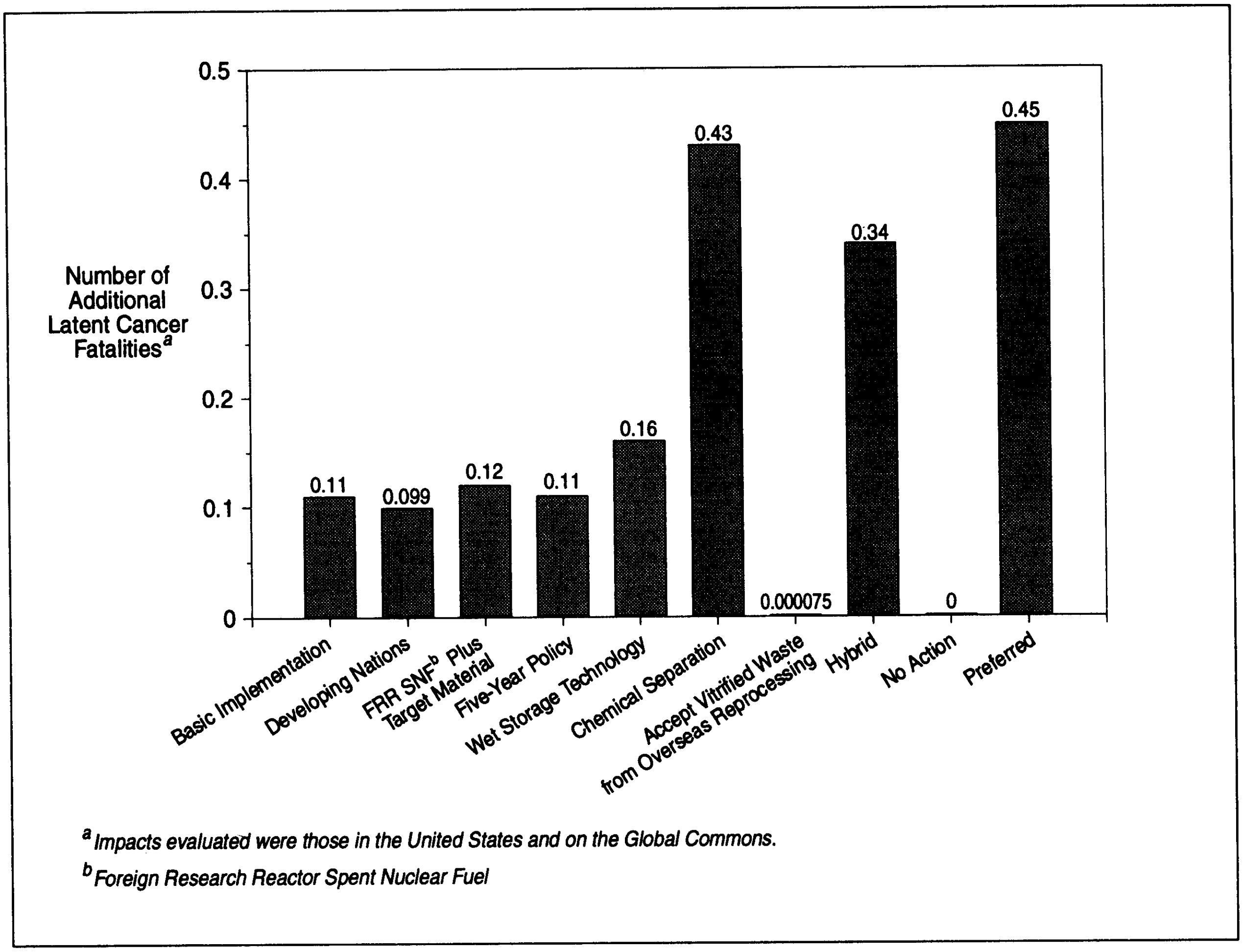


Figure 4-26 Maximum Estimated Accident Radiological Population Risk to the General Public Under Each Alternative

amount of material than identified in the basic implementation of Management Alternative 1. As mentioned above, additional storage space, if required, would be created on DOE-owned land, creating no outside land use issues.

Acceptance of foreign research reactor spent nuclear fuel for periods of time different from the time periods identified in the basic implementation of Management Alternative 1 would not cause any land use issues as the timeframe would not necessarily change the amount of foreign research reactor spent nuclear fuel received by the United States. If a policy of 5 years of acceptance was instituted, less spent nuclear fuel would be received by the United States, and if an indefinite HEU/10-year LEU policy were to be adopted, storage space would be created on DOE management sites, causing no issues in relation to outside lands.

Implementation through financial arrangements different from those identified in the basic implementation of Management Alternative 1 would have no impact on land use, as this alternative would have no effect on lands not owned by DOE.

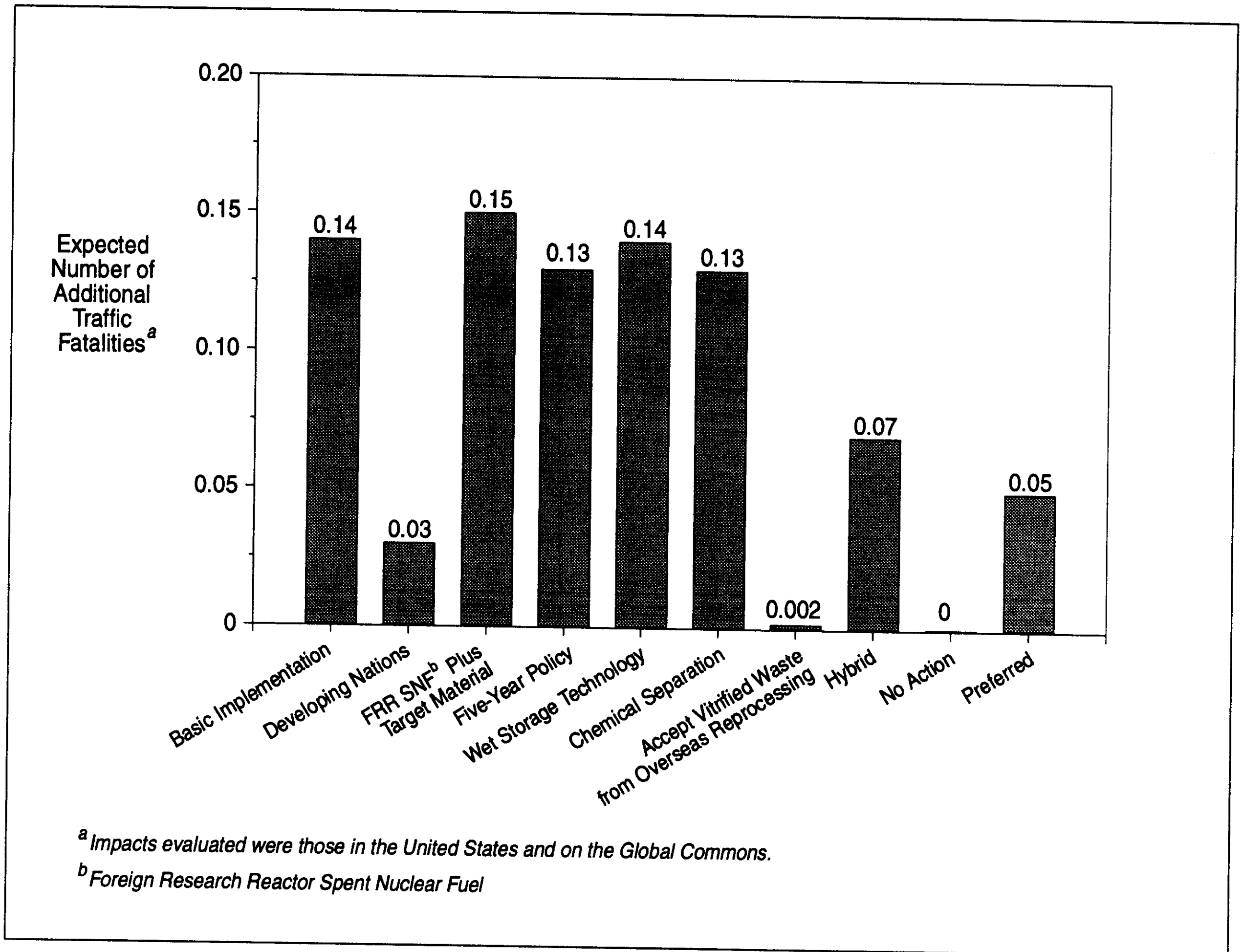


Figure 4-27 Maximum Estimated Traffic Accident Risk Under Each Alternative

Implementation by taking title to the foreign research reactor spent nuclear fuel at locations different from those identified in the basic implementation of Management Alternative 1 would cause no land use issues, as it would have no effect on the storage needs or the amount of foreign research reactor spent nuclear fuel received by the United States.

Use of wet storage technology for the interim period instead of dry storage technology as identified in the basic implementation of Management Alternative 1 would cause no land use issues, as the storage facilities (wet or dry) would be on DOE-owned land, and would have no effect on outside (non-DOE-owned) lands. If DOE decides to purchase the BNFP facility for interim wet storage, however, this would require adding some land to the Savannah River Site.

Implementation by use of near term chemical separation in the United States instead of interim storage would have no impact on land use, as the separation would be performed on DOE-owned land, with no effect on outside (non-DOE-owned) lands.

Similarly, there would be no land use concerns under either of the overseas subalternatives or the Hybrid Alternative presented in this EIS. A policy of no action (the No Action Alternative) regarding foreign research reactor spent nuclear fuel would cause no land use issues in the United States.

Land use for construction under the preferred alternative would be similar to the land use for construction under the basic implementation of Management Alternative 1.

4.8.6 Cultural Resources

Basic Implementation of Management Alternative 1: The basic implementation of Management Alternative 1 would not result in major impact to the cultural resources of the management sites being considered for the storage of the foreign research reactor spent nuclear fuel. Although the sites have not been evaluated and audited for cultural resources, surveys would be completed prior to any construction or other activity that would potentially disturb these areas. Areas of cultural or historical significance are the United States, the aluminum-based portion would be managed at the Savannah River Site and the TRIGA portion would be managed at the Idaho National Engineering Laboratory. DOE selected six scenarios, including the preferred alternative, for cost analysis. The costs of disposal were estimated for each scenario and are included in the analysis. The cost analysis also considers the financing arrangements discussed in Sections 2.2.1.2 and 2.2.2.3 that would affect the cost to the United States.

All costs are presented in two parts: 1) minimum discounted costs (base case) for the well-defined program components and integration approaches, and 2) "other cost factors" that are likely but sufficiently uncertain that they cannot be directly included in the minimum discounted costs. For the preferred alternative, however, a wide range of costs is presented because of the uncertainty associated with the new technology development program. An example of an item covered by "other cost factors" would be the cost growth caused by adverse weather that extends the time required to make shipments of the foreign research reactor spent nuclear fuel. The costs are shown as net present values in a consistent accounting framework.

4.9.1 Scenarios Analyzed

For the purpose of the cost analysis, six scenarios were analyzed. The scenarios reflect the alternatives that affect cost directly, and are consistent with the Record of Decision of the Programmatic SNF&INEL Final EIS (DOE, 1995c). The six cost scenarios are:

1. *Management Alternative 1 (Storage)* — Storage of aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site in new dry or wet storage facilities; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory in existing wet or dry storage facilities.

2. *Management Alternative 1 (revised to incorporate chemical separation)* — Chemical separation of aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

3. *Target Material* — Storage of target material at the Savannah River Site. This scenario provides the cost differential that can be used to assess the cost of managing target material in addition to the foreign research reactor spent nuclear fuel in Management Alternative 1 storage and chemical separation scenarios.

4. *Management Alternative 2* — Management of all foreign research reactor spent nuclear fuel overseas. This scenario reflects a combination of reprocessing and dry storage overseas. Countries with the capability to accept the waste from reprocessing are assumed to have their spent nuclear fuel reprocessed. The rest use dry storage.

5. *Management Alternative 3* — Chemical separation of a portion of the aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site; reprocessing of the remainder of aluminum-based foreign research reactor spent nuclear fuel overseas; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

6. *Preferred Alternative* - Implementation of a new treatment and/or packaging technology for aluminum-based foreign research reactor spent nuclear fuel and target material at the Savannah River Site; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

By varying the quantities of material managed in different ways in the United States and overseas, different cost scenarios can be generated. The costs of these variations are bounded by the costs of the scenarios described above. For instance, a management alternative that includes acceptance of target material into the United States would be represented by a combination of Scenarios 1 and 3 or 2 and 3.

The implementation alternatives under Management Alternative 1 related to alternative amounts of foreign research reactor spent nuclear fuel eligible under the policy (Section 2.2.2.1), and alternative policy durations (Section 2.2.2.2), were not considered separately in the cost analysis because they are bounded by the cost scenarios analyzed. These implementation alternatives reduce the amount of foreign research reactor spent nuclear fuel eligible under the policy.

The implementation alternative under Management Alternative 1 related to alternative locations for taking title to the foreign research reactor spent nuclear fuel (Section 2.2.2.4) was not considered because it does not affect the cost analysis.

4.9.2 Minimum Program Costs

Table 4-64 shows the minimum discounted program costs (base case) for the six scenarios defined above. These costs cover all foreign research reactor spent nuclear fuel shipments, management over 40 years, and geologic disposal. Uncertainties (risks) and escalation are zero. Costs to manage target material (Scenario 3) could be added to the costs of Scenarios 1, 2, 4, and 5 to produce a minimum program cost. Costs to manage target material are included in the preferred alternative (Scenario 6).

Table 4-64 Minimum Program Costs (Net Present Value, Millions of 1996 Dollars in 1996)

Scenario	*Net Present Value*
1. Management Alternative 1 (Storage)	725/775[a]
2. Management Alternative 1 (revised to incorporate Chemical Separation)	625
3. Target Material	35
4. Management Alternative 2	1,250
5. Management Alternative 3	675
6. Preferred Alternative[b]	625-950

[a] *Dry/Wet new storage facilities*

[b] *Includes target material*

The schedule for activities in Europe under Scenario 5 is similar to that in the United States but not exactly the same. Reprocessing takes place over 13 years at Dounreay (the same timespan used for chemical separation at the Savannah River Site) although it could be completed at Dounreay in 9 or 10 years. Dounreay's charges for reprocessing are based on 1996 costs, not costs for 1996 through 2008 averaged over the 13 year period (as was done for the Savannah River Site). Geologic disposal takes place in 2025 through 2030 in Europe and 2030 through 2035 in the United States.

Costs are discounted at 3 percent for the portion to be managed overseas and at 4.9 percent for the portion to be managed in the United States. These net present values imply that all funds required to pay for the program over its 40-year life are received and placed in a trust fund accruing interest at a 4.9 percent real rate of return. This rate of return is required by the Office of Management and Budget for the year ending February, 1996.

Because of the uncertainties involved with the implementation of the new technology, the cost for Scenario 6 (preferred alternative) is presented as a range as discussed in Appendix F, Section F.7.2.9. Also, the shipping costs in Scenario 6 include the assumption that only 38 cask shipments would be accepted on the West Coast.

4.9.3 Other Cost Factors

There are four important sources of cost risk (excluding escalation) that are not part of the minimum costs in Table 4-64. Table 4-65 shows the likely values (risks) for these factors, taking into account the absolute values of the uncertainties and their probability of occurrence.

Table 4-65 Other Cost Factors (Net Present Value, Millions of 1996 Dollars in 1996)

Scenario	*Cost Factors*				
	Systems Integration & Logistics	*Component Risks*	*Non-program Risks*	*3% Discount Rate*	*Range*
1. Management Alternative 1 (Storage)	100	75	35	175	385
2. Management Alternative 1 (revised to incorporate Chemical Separation)	100	±15	10	125	200-250
3. Target Material	5	5	0	25	35
4. Management Alternative 2	100	±500	1000	250	350-1850
5. Management Alternative 3	100	±10	150	75	315-335
6. Preferred Alternative[b,c]	100	75	35	225	435

[a] *It is assumed that risks are the same for dry or wet storage options.*

[b] *Includes target material*

[c] *It is assumed that risk factors are the same as Management Alternative 1 (Storage)*

The other cost factors summarized in Table 4-65 are as follows:

1. *Systems Integration and Logistics Risks* - Significant risks exist in the details of the policy implementation. The implementation of the policy would involve up to 41 foreign countries, 13 years of possible receipts, dozens of foreign ports, up to ten domestic ports, two U.S. management sites, and possibly several new facilities. Technical and procedural bottlenecks could arise in several areas.

2. *Component Risks* - Significant risks exist for specific components of the foreign research reactor spent nuclear fuel program, e.g., the comprehensiveness of the acceptance criteria for aluminum-clad spent nuclear fuel characterization for dry storage, the methods of spent nuclear fuel disposal, the cost allocation at existing and new facilities, and development of new technology.

3. *Non-Program Risks* - Significant risks exist for components of other programs that affect the implementation of the foreign research reactor spent nuclear fuel EIS, (e.g., escalating repository costs, adoption of monitored retrievable storage, and differences in facility utilization plans between this EIS and those of other EISs affecting the Savannah River Site and the Idaho National Engineering Laboratory). For Scenario 5, the risks are that no spent nuclear fuel infrastructure exists in more than half of the eligible countries and that no geologic disposal program exists in most of the eligible countries.

4. *Discount Rate Risks* - Significant risks exist that the current discount rate required by the

The program costs presented in Tables 4-64, 4-65, and 4-66 are in constant 1996 dollars, discounted to 1996. This implies that funds required to cover these costs are received in 1996 and [illegible] placed in a trust fund. If payments into the [illegible] had been [illegible]

Table 4-67 Costs to the United States for Minimum Program Under Various Scenarios and Fee Structures (Millions of 1996 Dollars, Net Present Value of Costs in 1996, Fees Levelized over 1996-2008 Period)

Scenario[a]	Full-Cost Recovery[b]	Levelized Shipping Fee $/kgTM	Levelized Management Fee (excluding shipping) $/kgTM	Net Present Value For Levelized Fee[c] (developed countries only)				No Fee[d]	Total (excluding shipping)
				$2,000/ kgTM	$5,000/ kgTM	$7,500/ kgTM	$10,000/ kgTM	Developed Countries	
1. Management Alternative 1 (Storage)	100	1,500	6,500	325	100	(75)	(250)	475	575
2. Management Alternative 1 (revised to incorporate Chemical Separation)	90	1,500	5,800	275	50	(125)	(300)	425	525
4. Management Alternative 2[f]	500+							1,250 +	1,750+
5. Management Alternative 3[g]	85	1,500	6,000	225	75	(50)	(175)	300	375
6. Preferred Alternative[e]	90-110	1,700	5,600-9,200	275-550	50-325	(150)-125	(325)-(50)	425-700	500-800

[a] *The total mass (kgTM) of foreign research reactor spent nuclear fuel in the various scenarios is approximately as follows: Aluminum-based plus TRIGA: 115,000 kgTM; from other-than-high-income-economy countries: 15,000 kgTM; from high-income-economy countries: 100,000kgTM; to Dounreay in Scenario 5: 37,000 kgTM. The total mass of target material is approximately 3,400 kgTM, essentially all from high-income-economy countries.*

[b] *Full-cost recovery from high-income-economy countries only. The United States bears the costs of the other-than-high-income-economy countries in these cases.*

[c] *Payable in 13 equal annual installments on December 31 of the years 1996 through 2008. Add costs in column labeled "Full-Cost Recovery" to generate total cost to the United States.*

[d] *As above, implicitly paid by the taxpayers in 13 equal annual installments (to maintain consistency with the payment period of the reactor operators), excluding shipping. The net present value of shipping in Scenarios 1 and 2 is $140 Million. The net present value of shipping to the U.S. only in Scenario 5 is $90 Million. The net present value of shipping in Scenario 6 is $160 million. Adding shipping to the net present value for Scenario 2 and Scenario 5 shows that the total program costs for Scenario 5 are slightly lower.*

[e] *Includes target material*

[f] *There is no defined basis for the charges to the United States for non-U.S. management. Costs to the United States under Scenario 4 assume that the United States absorbs the cost to construct and operate independent foreign research reactor spent nuclear fuel storage installations (including all supporting safety, security, transport, health physics, etc. infrastructure) for the 22 countries with no commercial nuclear power programs and that the United States partially subsidizes the other countries, depending on their income-economy status, commercial nuclear power infrastructure, and other factors.*

[g] *Revenues paid to the United States include pass-through of shipping charges. Costs to the United States for management in Europe include the cost of blending down the HEU to LEU ($20 million).*

Table 4-67 shows that for minimum discounted program costs and fees charged to high-income-economy country reactor operators levelized over 13 years, costs to the United States for the scenarios could range from several hundred million dollars at a fee of $2,000/kgTM to a profit for fees of $7,500/kgTM to $10,000/kgTM. The cost of managing the spent nuclear fuel from the other-than-high-income-economy countries (including shipping) adds roughly $100 million more to the cost borne by the United States.

If fees in the $2,000 to $10,000 per kgTM range are established and charged over 13 years, the costs to the United States would be as estimated in Table 4-67 plus any additional cost factors not incorporated in the minimum program costs. These additional cost factors are: 1) technical risks, 2) discount rate-related risks, and 3) escalation. Table 4-66 shows that technical risks could add roughly $100 to $200 million to the costs borne by the United States. Discount rate-related risks are of a similar size. Escalation risks are uncertain but could be in the same range.

4.10 Foreign Research Reactor Spent Nuclear Fuel Risks and Common Risks

This section compares foreign research reactor spent nuclear fuel program risks to those of common activities, such as smoking, flying, receiving a medical X-ray, and so forth.

4.10.1 Risks in the Proposed Action

Preceding sections in Chapter 4 evaluated the risks from radiological and nonradiological activities and accidents in four segments: marine transport, port activities, ground transport, and site activities.

The highest estimated accident MEI risk to the general public from any of the foreign research reactor spent nuclear fuel implementation alternatives is 0.00015 LCF, as shown earlier in Figure 4-23. This would be an individual who lives at the Oak Ridge Reservation boundary under Implementation Alternative 5, Wet Storage Technology for New Construction. This hypothetical individual's chance of incurring a fatal cancer would be increased by less than two in ten thousand

The highest estimated incident-free population risk to the general public living near any of the DOE management sites from any of the implementation alternatives is less than one-half LCF, as shown earlier in Figure 4-24. This risk occurs under Implementation Alternative 6, Near Term Chemical Separation in the United States, at the Savannah River Site. This risk would be spread among the roughly 600,000 people who live within 80 km (50 miles) of the Savannah River Site, so the average risk among these people would be less than one in a million.

The population risk to the general public due to radiation exposure during ground transport could be as high as 0.22 LCF, as discussed earlier under several of the implementation alternatives to Management Alternative 1.

Nonradiological fatalities are also unlikely. As a practical matter, the only source of nonradiological fatalities to the public is through a traffic accident with a truck or a train. Since truck or train shipments are about 100 or fewer per year, the likelihood of a crash is not high.

4.10.2 Common Radiological Risks

Table 4-68 presents several typical sources of exposure to radiation from everyday life (DOE, 1993e). The average person in the United States receives about 300 mrem each year from natural sources of radiation and about another 50 mrem from manmade sources of radiation. For example, the largest dose listed in Table 4-68 is the 200 mrem/yr from exposure to naturally-occurring radon gas. This is twice the

100 mrem/yr regulatory limit that would apply to marine workers, port workers, and truck drivers under the proposed action. It is also much higher than the dose any member of the general public would be likely to receive.

Table 4-68 Typical Sources of Radiation, Exposures, and Risks

Source	Dose Rate (mrem/yr)	Risk (LCF/yr)

5. Applicable Laws, Regulations, and Other Requirements

5.1 Consultation

Certain Federal laws, such as the Endangered Species Act, the Fish and Wildlife Coordination Act, and the National Historic Preservation Act, require consultation and coordination by the United States Department of Energy (DOE) with other governmental entities. These consultation and coordination requirements will commence and be completed as site-specific spent nuclear fuel management projects and decisions are proposed. Any site-specific required consultations will be addressed in the site-specific Environmental Impact Statement (EIS) and/or in Volume I of DOE Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Draft EIS (Table 5-1).

5.2 Laws and Other Requirements

This section identifies and summarizes the major laws, regulations, Executive Orders, and DOE Orders that may apply to the receipt and management of spent nuclear fuel from foreign research reactors.

Section 5.2.1 discusses the major Federal statutes that impose environmental protection and compliance requirements upon DOE. In addition, there may be other State and local measures applicable to the foreign research reactor spent nuclear fuel because Federal law delegates enforcement or implementation authority to State or local agencies. These state- and local-specific requirements are addressed in the site specific appendices. Section 5.2.2 addresses [illegible]

[illegible] issues of national policy and set guidelines under which Federal agencies, including DOE, must act. DOE implements its responsibilities for protection of public health, safety, and the environment through a series of Departmental Orders that are mandatory for operating contractors of DOE-owned facilities. Section 5.2.3 discusses those DOE orders related to environmental, health, and safety protection. Hazardous and radioactive materials transportation regulations are summarized in Section 5.4.2.

Table 5-1 Agency Consultations

Subject Area	Legislation	Agency
Endangered Species	Endangered Species Act of 1973, as amended; State laws	U.S. Fish and Wildlife Service, State agencies
Migratory birds	Migratory Bird Treaty Act	U.S. Fish and Wildlife Service
Bald and Golden eagles	Bald and Golden Eagle Protection Act	U.S. Fish and Wildlife Service
Archaeological, historical, and cultural preservation	National Historic Preservation Act of 1966, Archaeological Resources Protection Act, Antiquities Act, American Indian Religious Freedom Act of 1978, Native American Grave Protection and Repatriation Act of 1990	State Historic Preservation Office, President's Advisory Council, Tribes
Discharge of pollutants to water	Clean Water Act, Safe Drinking Water Act	U.S. Environmental Protection Agency, State agencies
Work in navigable U.S. waters	Clean Water Act, Rivers and Harbors Act, Coastal Management Act	U.S. Army Corps of Engineers
Prime and unique farmlands	Farmland Protection Policy Act of 1981	Soil Conservation Service
Floodplains	Executive Order 11988, Fish and Wildlife Coordination Act	U.S. Army Corps of Engineers, U.S. Fish and Wildlife Service, State agencies
Wetlands	Executive Order 11990, Fish and Wildlife Coordination Act, Clean Water Act	U.S. Army Corps of Engineers, U.S. Fish and Wildlife Service, U.S. Environmental Protection Agency, State agencies
Environmental justice	Executive Order 12898	U.S. Environmental Protection Agency
Water body alteration	Fish and Wildlife Coordination Act	U.S. Fish and Wildlife Service, State agencies
River status	Wild and Scenic Rivers Act, Anadromous Fish Conservation Act, Hanford Reach Study Act	U.S. Department of the Interior
Air pollution	Clean Air Act	U.S. Environmental Protection Agency, State and local agencies
Water use and availability	Water Resources Planning Act of 1965, Safe Drinking Water Act, and others	U.S. Environmental Protection Agency, Office of Water Policy, State agencies
Noise	Noise Pollution and Abatement Act of 1970, Noise Control Act of 1972	U.S. Environmental Protection Agency, State agencies
Siting and planning	State siting acts, county zoning regulations	State and County agencies
Waste management and transportation	Solid Waste Disposal Act, as amended by the Resource Conservation and Recovery Act and the Hazardous and Solid Waste Amendments of 1984; Comprehensive Environmental Response, Compensation, and Liability Act; Emergency Planning and Community Right to Know Act; Hazardous Materials Transportation Act	U.S. Environmental Protection Agency, U.S. Department of Transportation, U.S. Coast Guard, State agencies
Emergency Management & Response	Defense Production Act of 1950, Robert T. Stafford Disaster Relief and Emergency, Assistance Act, National Security Act of 1947	Federal Emergency Management Agency, U.S. Environmental Protection Agency, U.S. Department of Transportation, U.S. Coast Guard, State and local agencies

Atomic Energy Act of 1954, as amended (42 USC §2011 et seq.)

The Atomic Energy Act of 1954 authorizes DOE to establish standards to protect health or minimize dangers to life or property with respect to activities under its jurisdiction. Through a series of DOE Orders, DOE has established an extensive system of standards and requirements to ensure safe operation of its facilities.

Clean Air Act, as amended (42 USC §7401 et seq.)

The Clean Air Act, as amended, is intended to "protect and enhance the quality of the Nation's air resources so as to promote the public health and welfare and the productive capacity of its population." Section 118 of the Clean Air Act, as amended, requires that each Federal agency, such as DOE, with jurisdiction over any property or facility that might result in the discharge of air pollutants, comply with "all Federal, State, interstate, and local requirements" with regard to the control and abatement of air pollution.

The Act requires the Environmental Protection Agency to establish National Ambient Air Quality Standards as necessary to protect public health, with an adequate margin of safety, from any known or anticipated adverse effects of a regulated pollutant (42 USC §7409). The Act also requires establishment of national standards of performance for new or modified stationary sources of atmospheric pollutants (42 USC §7411) and requires specific emission increases to be evaluated so as to prevent a significant deterioration in air quality (42 USC §7470). Hazardous air pollutants, including radionuclides, are regulated separately (42 USC §7412). Air emissions are regulated by the Environmental Protection Agency in 40 CFR Parts 50 through 99. In particular, radionuclide emissions are regulated under the National Emission Standard for Hazardous Air Pollutants Program (see 40 CFR Part 61).

Safe Drinking Water Act, as amended [42 USC §300 (F) et seq.]

The primary objective of the Safe Drinking Water Act, as amended, is to protect the quality of the public water supplies and all sources of drinking water. The implementing regulations, administered by the Environmental Protection Agency unless delegated to the States, establish standards applicable to public water systems. They promulgate maximum contaminant levels (including those for radioactivity), in public water systems, which are defined as water systems that serve at least 15 service connections used by year-round residents or regularly serve at least 25 year-round residents. Safe Drinking Water Act requirements have been promulgated by the Environmental Protection Agency in 40 CFR Parts 100 through 149. For radioactive material, the regulations specify that the average annual concentration of manmade radionuclides in drinking water as delivered to the user by such a system shall not produce a dose equivalent to the total body or an internal organ greater than four mrem per year beta activity. Other programs established by the Safe Drinking Water Act include the Sole Source Aquifer Program, the Wellhead Protection Program, and the Underground Injection Control Program.

Clean Water Act, as amended (33 USC §1251 et seq.)

The Clean Water Act, which amended the Federal Water Pollution Control Act, was enacted to "restore and maintain the chemical, physical and biological integrity of the Nation's water." The Clean Water Act prohibits the "discharge of toxic pollutants in toxic amounts" to navigable waters of the United States. Section 313 of the Clean Water Act, as amended, requires all branches of the Federal Government engaged in any activity that might result in a discharge or runoff of pollutants to surface waters to comply with Federal, State, interstate, and local requirements.

In addition to setting water quality standards for the Nation's waterways, the Clean Water Act supplies guidelines and limitations for effluent discharges from point-source discharges and provides authority for the Environmental Protection Agency to implement the National Pollutant Discharge Elimination System permitting program. The National Pollutant Discharge Elimination System program is administered by the Water Management Division of the Environmental Protection Agency pursuant to regulations in 40 CFR Part 122 et seq. Idaho has not applied for National Pollutant Discharge Elimination System authority from the Environmental Protection Agency. Thus, all National Pollutant Discharge Elimination System permits required for the Idaho National Engineering Laboratory are obtained by DOE through Environmental Protection Agency Region 10 (40 CFR Part 122 et seq.).

Sections 401 and 405 of the Water Quality Act of 1987 added Section 402(p) to the Clean Water Act. Section 402(p) requires that the Environmental Protection Agency establish regulations for issuing permits for stormwater discharges associated with industrial activity. Although any stormwater discharge associated with industrial activity requires a National Pollutant Discharge Elimination System permit application, regulations implementing a separate stormwater permit application process have not yet been adopted by the Environmental Protection Agency.

Resource Conservation and Recovery Act, as amended (Solid Waste Disposal Act) (42 USC §6901 et seq.)

The treatment, storage, or disposal of hazardous and nonhazardous waste is regulated under the Solid Waste Disposal Act as amended by the Resource Conservation and Recovery Act and the Hazardous and Solid Waste Amendments of 1984. Pursuant to Section 3006 of the Act, any State that seeks to administer and enforce a hazardous waste program pursuant to the Resource Conservation and Recovery Act may apply for Environmental Protection Agency authorization of its program. The Environmental Protection Agency regulations implementing the Resource Conservation and Recovery Act are found in 40 CFR Parts 260 through 280. These regulations define hazardous wastes and specify hazardous waste transportation, handling, treatment, storage, and disposal requirements.

The regulations imposed on a generator or a treatment, storage, and/or disposal facility vary according to the type and quantity of material or waste generated, treated, stored, and/or disposed of. The method of treatment, storage, and/or disposal also impacts the extent and complexity of the requirements.

Current Status of Spent Nuclear Fuel under the Resource Conservation and Recovery Act

Historically, DOE chemically reprocessed spent nuclear fuel to recover valuable products and fissionable materials, and as such, the spent nuclear fuel was not a solid waste under the Resource Conservation and Recovery Act.

World events have resulted in significant changes in DOE's direction and operations. In particular, in April 1992, DOE announced the phase-out of reprocessing for the recovery of special nuclear materials. With these changes, DOE's focus on most of its spent nuclear fuel has changed from reprocessing and recovery of materials to storage and ultimate disposition. This in turn has created uncertainty regarding the regulatory status of some of DOE's spent nuclear fuel relative to the Resource Conservation and Recovery Act.

DOE has initiated discussion with the Environmental Protection Agency on the potential applicability of the Resource Conservation and Recovery Act to spent nuclear fuel. Further discussions with Environmental Protection Agency Headquarters and regional offices and State regulators are ongoing to develop a strategy for meeting any the Resource Conservation and Recovery Act requirements that might apply.

Federal Facility Compliance Act (42 USC §6921 et seq.)

The Federal Facility Compliance Act, enacted on October 6, 1992, waives sovereign immunity for fines and penalties for Resource Conservation Recovery Act violations at Federal facilities. However, a provision postpones fines and penalties after 3 years for mixed waste storage prohibition violations at waste stored or generated at each facility. Each plan must be approved by the host State or the Environmental Protection Agency, after consultation with other affected States, and a consent order must be issued by the regulator requiring compliance with the plan. The Federal Facility Compliance Act further provides that DOE will not be subject to fines and penalties for land disposal restriction storage prohibition violations for mixed waste as long as it is in compliance with such an approved plan and consent order and meets all other applicable regulations. This would only apply to foreign research reactor spent nuclear fuel if the Resource Conservation and Recovery Act would apply to storage and treatment of foreign research reactor spent nuclear fuel.

National Historic Preservation Act, as amended (16 USC §470 et seq.)

The National Historic Preservation Act, as amended, provides that sites with significant national historic value be placed on the *National Register of Historic Places*. There are no permits or certifications required under the Act. However, if a particular Federal activity may impact a historic property resource, consultation with the Advisory Council on Historic Preservation will usually generate a Memorandum of Agreement, including stipulations that must be followed to minimize adverse impacts. Coordination with the State Historic Preservation officer is also undertaken to ensure that potentially significant sites are properly identified and appropriate mitigative actions are implemented.

Religious Freedom Restoration Act of 1993 (42 USC §2000bb et seq.)

This Act prohibits the Government, including Federal Departments, from substantially burdening the exercise of religion unless the Government demonstrates a compelling governmental interest, and the action furthers a compelling Government interest and is the least restrictive means of furthering that interest.

Endangered Species Act, as amended (16 USC §1531 et seq.)

The Endangered Species Act, as amended, is intended to prevent the further decline of endangered and threatened species and to restore these species and their habitats. The Act is jointly administered by the United States Departments of Commerce and the Interior. Section 7 of the Act requires consultation with the U.S. Fish and Wildlife Service to determine whether endangered and threatened species or their critical habitats are known to be in the vicinity of the proposed action. The Idaho National Engineering Laboratory has commenced the consultation process with the U.S. Fish and Wildlife Service (DOE, 1995c). The Savannah River Site, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site have also commenced consultations with the U.S. Fish and Wildlife Service.

Migratory Bird Treaty Act, as amended (16 USC §703 et seq.)

The Migratory Bird Treaty Act, as amended, is intended to protect birds that have common migration patterns between the United States and Canada, Mexico, Japan, and Russia. It regulates the harvest of migratory birds by specifying things such as the mode of harvest, hunting seasons, and bag limits. The Act stipulates that it is unlawful at any time, by any means, or in any manner to "kill . . . any migratory bird." Although no permit for this project is required under the Act, DOE is required to consult with the U.S. Fish and Wildlife Service regarding impacts to migratory birds and to evaluate ways to avoid or minimize these effects in accordance with the U.S. Fish and Wildlife Service Mitigation Policy.

Bald and Golden Eagle Protection Act, as amended (16 USC §668-668d)

The Bald and Golden Eagle Protection Act makes it unlawful to take, pursue, molest, or disturb bald (American) and golden eagles, their nests, or their eggs anywhere in the United States (Sections 668, 668c). A permit must be obtained from the U.S. Department of the Interior to relocate a nest that interferes with resource development or recovery operations.

Wild and Scenic Rivers Act, as amended (16 USC 1271 et seq. 71:8301 et seq.)

The Wild and Scenic Rivers Act, as amended, protects certain selected rivers of the Nation that possess outstanding scenic, recreational, geological, fish and wildlife, historical, cultural, or other similar values. These rivers are to be preserved in a free-flowing condition to protect water quality and other vital national conservation purposes. The purpose of the Act is to institute a national wild and scenic rivers system, to designate the initial rivers that are a part of that system, and to develop standards for the addition of new rivers in the future.

Occupational Safety and Health Act of 1970, as amended (29 USC §651 et seq.)

The Occupational Safety and Health Act establishes standards to enhance safe and healthful working conditions in places of employment throughout the United States. The Act is administered and enforced by the Occupational Safety and Health Administration, a U.S. Department of Labor agency. While the Occupational Safety and Health Administration and Environmental Protection Agency both have a mandate to reduce exposures to toxic substances, the Occupational Safety and Health Administration's jurisdiction is limited to safety and health conditions that exist in the workplace environment. In general, under the Act, it is the duty of each employer to furnish all employees a place of employment free of recognized hazards likely to cause death or serious physical harm. Employees have a duty to comply with the occupational safety and health standards and all rules, regulations, and orders issued under the Act. the Occupational Safety and Health Administration regulations (29 CFR) establish specific standards telling employers what must be done to achieve a safe and healthful working environment. DOE places emphasis on compliance with these regulations at its facilities and prescribes through DOE Orders the Occupational Safety and Health Act standards that contractors shall meet, as applicable to their work at Government-owned, contractor-operated facilities (DOE Order 5480.1B, 5483.1A). DOE keeps and makes available the various records of minor illnesses, injuries, and work-related deaths as required by the Occupational Safety and Health Administration regulations.

Noise Control Act of 1972, as amended (42 USC §4901 et seq.)

Section 4 of the Noise Control Act of 1972, as amended, directs all Federal agencies to carry out "to the fullest extent within their authority" programs within their jurisdictions in a manner that furthers a national policy of promoting an environment free from noise that jeopardizes health and welfare.

5.2.2 Executive Orders

Executive Order 11514 (Protection and Enhancement of Environmental Quality)

Executive Order 11514 requires Federal agencies to continually monitor and control their activities to protect and enhance the quality of the environment and to develop procedures to ensure the fullest practicable provision of timely public information and understanding of the Federal plans and programs with environmental impact to obtain the views of interested parties. The DOE has issued regulations (10 CFR 1021) and DOE Order 5440.1E for compliance with this Executive Order.

Executive Order 11988 (Floodplain Management)

Executive Order 11988 requires Federal agencies to establish procedures to ensure that the potential effects of flood hazards and floodplain management are considered for any action undertaken in a floodplain and that floodplain impacts be avoided to the extent practicable.

Executive Order 11990 (Protection of Wetlands)

Executive Order 11990 requires Governmental agencies to avoid any short- and long-term adverse impacts on wetlands wherever there is a practicable alternative.

Executive Order 12856 (Right-to-Know Laws and Pollution Prevention Requirements)

Executive Order 12856 requires all Federal agencies to reduce the toxic chemicals entering any waste stream. This order also requires Federal agencies to report toxic chemicals entering waste streams; improve emergency planning, response, and accident notification; and encourage clean technologies and testing of innovative prevention technologies.

Executive Order 12898 (Environmental Justice)

Executive Order 12898 requires Federal agencies to identify and address disproportionately high and adverse human health or environmental effects of its programs, policies, and activities on minority and low-income populations.

Table 5-2 DOE Orders Relevant to the DOE Spent Nuclear Fuel Management Program

DOE Order	Subject
1300.2A	Department of Energy Technical Standards Program (5-19-92)
1360.2B	Unclassified Computer Security Program (5-18-92)
1540.2	Hazardous Material Packaging for Transport-Administrative Procedures (9-30-86; Chg. 1. 12-19-88)
3790.1B	Federal Employee Occupational Safety and Health Program (1-7-93)
4330.4A	Maintenance Management Program (10-17-90)
4700.1	Project Management System (3-6-87)
5000.3B	Occurrence Reporting and Utilization of Operations Information (4-9-92)
5400.1	General Environmental Protection Program (11-9-88; Chg. 1, 6-29-90)
5400.2A	Environmental Compliance Issue Coordination (Errata 1-31-89)
5400.4	Comprehensive Environmental Response, Compensation, and Liability Act Requirements (10-6-89)
5400.5	Radiation Protection of the Public and the Environment (2-8-90; Chg. 2, 1-7-93)
5440.1E	National Environmental Policy Act Compliance Program (11-10-92)
5480.1B	Environmental, Safety and Health Program for DOE Operations (9-23-86; Chg. 4, 3-27-90)
5480.3	Environmental Requirements for the Packaging and Transportation of Hazardous Materials, Hazardous Substances, and Hazardous Wastes (7-9-85)
5480.4	Environmental Protection, Safety, and Health Protection Standards (5-15-84; Chg. 4, 1-7-93)
5480.6	Safety of Department of Energy-Owned Nuclear Reactors (9-23-86)
5480.7A	Fire Protection (2-17-93)
5480.8A	Contractor Occupational Medical Program (6-26-92)
5480.9	Construction Safety and Health Program (11-18-87)
5480.10	Contractor Industrial Hygiene Program (6-26-85)
5480.11	Radiation Protection for Occupational Workers (12-21-88; Chg. 2, 6-29-90)
5480.15	Department of Energy Laboratory Accreditation Program for Personnel Dosimetry (12-14-87)
5480.17	Site Safety Representatives (10-05-88))
5480.18A	Accreditation of Performance-Based Training for Category A Reactors and Nuclear Facilities (07-19-91)
5480.19	Conduct of Operations Requirements for DOE Facilities (7-9-90; Chg. 1, 5-18-92)
5480.20	Personnel Selection, Qualification, Training, and Staffing Requirements at DOE Reactor and Nonreactor Nuclear Facilities (2-20-91)
5480.21	Unreviewed Safety Questions (12-24-91)
5480.22	Technical Safety Requirements (2-25-92; Chg. 1, 9-15-92)
5480.23	Nuclear Safety Analysis Reports (4-10-92)
5480.24	Nuclear Criticality Safety (8-12-92)
5480.27	Equipment Qualification for Reactor and Nonreactor Nuclear Facilities (1-15-93)
5480.28	Natural Phenomena Hazards Mitigation (1-15-93)
5480.31	Startup and Restart of Nuclear Facilities (9-15-93)
5481.1B	Safety Analysis and Review System (9-23-86; Chg. 1, 5-19-87)
5482.1B	Environment, Safety, and Health Appraisal Program (9-23-86; Chg. 1, 11-18-91)
5483.1A	Occupational Safety and Health Program for DOE Contractor Employees at Government-Owned, Contractor-Operated Facilities (6-22-83)
5484.1	Environmental Protection, Safety, and Health Protection Information Reporting Requirements (2-21-81; Chg. 7, 10-17-90)
5500.1B	Emergency Management System (4-30-91; Chg. 1, 4-30-91)
5500.2B	Emergency Categories, Classes, and Notification and Reporting Requirements (4-30-91; Chg. 1, 2-27-92)
5500.3A	Planning and Preparedness for Operational Emergencies (4-30-91; Chg. 1, 2-27-92)
5500.4A	Public Affairs Policy and Planning Requirements for Emergencies (6-8-92)
5500.7B	Emergency Operating Records Protection Program (10-23-91)
5500.10	Emergency Readiness Assurance Program (4-30-91; Chg. 1, 2-27-92)
5530.3	Radiological Assistance Program (01-14-92; Change 1, 4-10-92)
5530.5	Federal Radiological Monitoring and Assessment Center (7-10-92)
5630.11A	Safeguards and Security Program (12-7-92)
5630.12A	Safeguards and Security Inspection and Evaluation Program (6-23-92)
5700.6C	Quality Assurance (8-21-91)
5820.2A	Radioactive Waste Management (9-26-88)
6430.1A	General Design Criteria (4-6-89)

5.2.3 DOE Regulations and Orders

Through the authority of the Atomic Energy Act, DOE is responsible for establishing a comprehensive health, safety, and environmental program for its facilities. The regulatory mechanisms through which DOE manages its facilities are the promulgation of regulations and the issuance of DOE Orders.

The DOE regulations are generally found in 10 CFR. These regulations address such areas as energy conservation, administrative requirements and procedures, nuclear safety, and classified information. For the purposes of this EIS, relevant regulations include 10 CFR Part 834, Radiation Protection of the Public and the Environment; 10 CFR Part 835, Occupational Radiation Protection; 10 CFR Part 1021, Compliance with NEPA; and 10 CFR Part 1022, Compliance with Floodplains/Wetlands Environmental Review Requirements. DOE has enacted occupational radiation protection standards to protect DOE and its contractor employees. These standards are set forth in 10 CFR Part 83b, Occupational Radiation Protection. The rules in this part establish radiation protection standards, limits, and program requirements for protecting individuals from ionizing radiation resulting from the conduct of DOE activities, including those conducted by DOE contractors. The activity may be, but is not limited to, design, construction, or operation of DOE facilities. These regulations would be in effect for the

The emphasis of the International Atomic Energy Agency model regulations is on package integrity. To that end, packagings must be shown to survive a hypothetical accident sequence that includes impact, crush, puncture, fire, and immersion. The level of protection is defined by the nature of the contents. The intent of the regulations is to maximize the shipper's contribution to safety, and the shipper (consignor) must certify "that the contents of this consignment are properly described by name; are properly packaged, marked and labeled; and are in proper condition for transport ... " (IAEA, 1990a). The carrier is responsible for following rules for stowage and for segregation from persons.

International Maritime Organization Regulations

The International Maritime Organization publishes the International Maritime Dangerous Goods Code (IMO, 1994), which was developed to supplement the provisions of the 1960 International Convention on the Safety of Life at Sea, as amended, (IMO, 1992) to which the United States is a signatory. Included are regulations that deal with carriage of radioactive material (Class 7 materials). They are based on the International Atomic Energy Agency regulations and deal with segregation of radioactive materials packages from other dangerous goods and other aspects of stowage.

5.4 Domestic Regulations for Radioactive Material Packaging and Transportation

Hazardous and Radioactive Materials Transportation Regulations

Transportation of hazardous and radioactive materials, substances, and wastes are governed by the Department of Transportation, NRC, and the Environmental Protection Agency regulations. These regulations may be found in 49 CFR Parts 171 through 178, 49 CFR Parts 383 through 397, 10 CFR Part 71, and 40 CFR Parts 262 and 265, respectively.

Department of Transportation regulations contain requirements for identifying a material as hazardous or radioactive. These regulations interface with NRC or the Environmental Protection Agency regulations for identifying material, but the Department of Transportation hazardous material regulations govern the hazard communication (such as marking, hazard labeling, vehicle placarding, and emergency response telephone number) and shipping requirements (such as required entries on shipping papers or the Environmental Protection Agency waste manifests).

NRC regulations applicable to radioactive materials transportation are found in 10 CFR Part 71, which includes detailed packaging design requirements and package certification testing requirements. Complete documentation of design and safety analysis and results of the required testing are submitted to the NRC to certify the package for use. This certification testing involves the following components: heat, physical drop onto an unyielding surface, water submersion, puncture by dropping package onto a steel bar, and gas tightness. The recent revision of 10 CFR Part 71, issued on September 28, 1995 (60 CFR 50248), is intended primarily to bring this regulation into conformance with current International Atomic Energy Agency regulations. Revised regulations applicable to the transportation of spent nuclear fuel from foreign research reactors are essentially unchanged.

The Environmental Protection Agency regulations pertaining to hazardous waste transportation are found in 40 CFR Parts 262 and 265. These regulations address labeling and record keeping requirements, including the use of the Environmental Protection Agency waste manifest, which is the required shipping paper for transporting the Resource Conservation and Recovery Act hazardous waste.

5.4.1 NRC Packaging Certification

An NRC certificate is issued as evidence that a packaging and its contents meet applicable Federal regulations. The certificate is issued on the basis of a Safety Analysis Report on the packaging design. Type B packaging must survive certain severe hypothetical accident conditions of impact, puncture, fire, and immersion. The tests are not intended to duplicate accident environments, but rather to produce damage equivalent to extreme accidents. The complete accident sequence is described in 10 CFR, Part 71.73.

Test Sequence for Type B Packagings

The effects of the tests on a package may be evaluated either by subjecting a scale model sample package to the test or by other methods acceptable to the NRC. NRC Regulatory Guide 7.9 allows assessment of package performance by analysis, prototype testing, model testing, or comparison to a similar package. To be judged as surviving, the packaging must not exceed allowable releases defined in 10 CFR 71.51. The dose rate outside the packaging must not exceed 1 rem per hour at a distance of 1 m (3.3 ft) from the packaging surface. The first three tests must be performed on the same package in this order: drop test, puncture test, and thermal test (with an immersion test following for fissile material packagings only).

The drop test consists of a 9-m (30-ft) drop onto a flat, essentially unyielding, horizontal surface, striking the package surface in the position for which maximum damage is expected. An essentially unyielding surface is one that absorbs very little of the energy of impact, which means that the energy of impact is absorbed almost entirely by the package. Unyielding surfaces are constructed of a monolithic concrete base, reinforced by Rebar and covered with a plate of battleship armor. The puncture test consists of a 1-m (40-in) drop onto the upper end of a 15-cm (6-in) solid, vertical, cylindrical bar of mild steel mounted on an essentially unyielding surface. The top of the bar must be horizontal and its edge rounded to a radius of not more than 6 mm (0.25 in).

In the thermal test, the packaging must be exposed for not less than 30 minutes to a heat flux not less than that of a radiation environment of 800°C (1,475°F) with an emissivity coefficient of a least 0.9. The surface absorptivity must be either the value that the package may be expected to possess if exposed to a fire or 0.8, whichever is greater. When it might be significant, convective heat input must be included on the basis of still, ambient air. The packaging may not be artificially cooled after external heat input ceases, and any combustion of materials of construction must be allowed to proceed until it terminates naturally.

Fissile materials packagings for which water in leakage has not been assumed for criticality analysis must be subjected to submersion under a head of water of at least 0.9 m (3 ft) for not less than 8 hours and in the attitude for which the maximum leakage is expected. All packages must be subjected to a separate test in which an undamaged cask is submerged under a head of water of at least 15 m (50 ft) for not less than 8 hours.

Although spent fuel casks have been involved in several accidents, their integrity has never been compromised. The regulatory tests are structured to place an upper bound on the kinds of damage seen in actual severe transportation accidents. Furthermore, after completion of this series of performance qualification tests, Type B packagings are further subjected to a post-accident leak-rate performance test (10 CFR 71.51). In this test, no escape of radioactive material is allowed that exceeds an A2 amount in a week. The A2 amount of an isotope is the maximum activity of that isotope in a potentially dispersible form that is allowed to be shipped in a Type A packaging, which is nonaccident resistant. Safety Series No. 6 lists A2 values for all commonly transported isotopes.

The NRC revised 10 CFR Part 7 regulations governing the transportation of radioactive materials on September 28, 1995 (60 FR 50248). These regulations become effective on April 1, 1996 (NRC, 1995). The revised regulations conform with those of the International Atomic Energy Agency and current legislative requirements. The revised regulations affecting "Type B" casks require that a spent nuclear fuel transportation cask with activity greater than 106 curies be designed and constructed so that its undamaged containment system would withstand an external water pressure of 290 psi, or immersion in 200 meters (656 ft) of water, for a period of not less than one hour without collapse, buckling, or allowing water to leak into the cask.

The use of an essentially unyielding target makes the regulatory certification tests extremely demanding. Real targets are much more yielding. For example, a lead-shield steel cask was dropped 610 m (2,000 ft) from a helicopter onto undisturbed soil (NRC, 1977). Impact velocity was 396 km per hour (235 mph)

Shipping papers should have entries identifying the following: the name of the shipper, emergency response telephone number, description of spent nuclear fuel, and the shipper's certificate as described in 49 CFR §172 Subpart C.

In addition, drivers of motor vehicles transporting spent nuclear fuel must have training in accordance with the requirements of 49 CFR §172.700. The training requirements include: familiarization with the regulations, emergency response information, and the spent nuclear fuel communication programs required by the Occupational Safety and Health Administration. Drivers are also required to have training on the procedures necessary for safe operation of the vehicle used to transport the spent nuclear fuel.

Except for exclusive-use shipments, requirements relating to transport indexes state that:

> ". . . the number of freight containers with packages of radioactive materials contained therein must be limited so that the total sum of the transport indexes in the containers in any hold or defined deck area does not exceed 200, and:
>
> (1) The sum of transport indexes for any individual freight container, or group of freight containers, does not exceed 50; and,
>
> (2) Each freight container or group of freight containers is (are) handled and stowed in such a manner that groups are separated from each other by a distance of at least six m (20 ft)," [49 CFR §176.704(c)].

Section 176.76(a) includes provision for freight containers with hazardous materials to be carried onboard a vessel in accordance with the following:

> " (1) The material must be in proper condition for transportation according to the requirements of this subchapter;
>
> (2) All packages in the transport vehicle or container must be secured to prevent movement in any direction. However, vertical restraint is not required if the shape of the packages and the stuffing pattern precludes shifting of the load;
>
> (3) Bulkheads made of dunnage which extend to the level of the cargo must be provided unless the packages are stowed flush with the sides or ends;
>
> (4) Dunnage must be secured to the floor when the cargo consists of dense materials or heavy packages."

Each freight container must be placarded as required by 49 CFR §172 Subpart F of the Hazardous Materials Regulations [49 CFR 176.76(f)].

Section 49 CFR 176.80 requires that radioactive materials be segregated from other hazardous materials so that they do not interact dangerously in an accident, or alternatively, requires that the radioactive material be in separate holds when stored under deck. In 49 CFR 176.83(b), a table is provided (Table II) that specifies the minimum separation distances for different classes of hazardous materials onboard a vessel. A minimum horizontal separation distance of 3 m (10 ft) projected vertically from the reference package is required. For specified hazardous materials, the "separate from" requirement means that the materials must be placed in separate holds when stowed under deck.

Marine Transport

Relevant regulations applying to transport of spent nuclear fuel by vessel are found in 10 CFR Parts 71 and 73, and 49 CFR Part 176. The USCG, part of the Department of Transportation, inspects vessels for compliance with applicable regulations and requires 24-hour prenotification (33 CFR 160.207, 211, and 213).

Section 49 CFR 171.12 (d) states that: "Radioactive materials being imported into or exported from the United States, or passing through the United States in the course of being shipped between places outside the U.S., may be offered and accepted for transportation when packaged, marked, labeled, and otherwise prepared for shipment in accordance with the IAEA 'Regulations for the Safe Transport of Radioactive Materials, Safety Series No. 6, 1985 Edition' including 'Supplement 1988.'" Certain specified conditions of this section must be complied with. For example, highway-route-controlled quantities of radioactive material must be shipped in accordance with appropriate provisions of the hazardous materials regulations and a Certificate of Competent Authority must be obtained, with any necessary revalidations. A Certificate of Competent Authority fulfills the International Atomic Energy Agency requirement for multilateral approval for a shipment of Type B packages in international commerce (IAEA, 1990a).

Section 49 CFR 176.5 details the application of the regulations to vessels: "...this subchapter applies to each domestic or foreign vessel when in the navigable waters of the United States, regardless of its character, tonnage, size or service, and whether self-propelled or not, whether arriving or departing, underway, moored, anchored, aground, or while in drydock." Exempted from the regulations are vessels not engaged in commercial service, a vessel used exclusively for pleasure, a vessel of 500 gross tons or smaller, engaged in fisheries, etc. Section 49 CFR 176.15 provides for enforcement of 40 CFR Subchapter C:

> "(a) An enforcement officer of the U.S. Coast Guard may at any time and at any place, within the jurisdiction of the United States, board any vessel for the purpose of enforcement of this subchapter and inspect any shipment of hazardous materials as defined in this subchapter."

Provision is also made in this section to detain a vessel that is in violation of the hazardous materials regulations.

The USCG may accept a certificate of loading issued by the National Cargo Bureau, Inc., as evidence that the cargo is stowed in conformity with law and regulatory requirements. The National Cargo Bureau, Inc., is a non-profit organization directed by government and industry representatives (49 CFR 176.18 authorizes inspectors of the National Cargo Bureau, Inc., to assist the USCG in administering the hazardous materials regulations). Their functions are as follows:

> "(1) Inspection of vessels for suitability for loading hazardous materials; (2) Examination of stowage of hazardous materials; (3) Making recommendations for stowage requirements of hazardous materials cargo; and, (4) Issuance of certificates of loading setting forth that the stowage of hazardous materials is in accordance with the requirements of 46 U.S.C. 170 and its subchapter."

Detailed requirements for shipping radioactive material are located in Part 176 Subpart M of the hazardous materials regulations. General radioactive materials stowage requirements of 49 CFR 176.700 state that: "(b) A package of radioactive materials which in still air has a surface temperature more than 5°C (9°F) above the ambient air may not be overstowed with any other cargo. If the package is stowed under the deck, the hold or compartment in which it is stowed must be ventilated."

Except for exclusive-use shipments, requirements of 176.704 (c) relating to transport indexes state that:

> "the number of freight containers with packages of radioactive materials contained therein must be limited so that the total sum of the transport indexes in the containers in any hold or defined deck area does not exceed 200, and: (1) The sum of transport indexes for any individual freight container, or group of freight containers, does not exceed 50; and, (2) Each freight container or group of freight containers is handled and stowed in such a manner that groups are separated from each other by a distance of at least six meters (20 feet)."

Section 176.76(a) includes provision for freight containers with hazardous materials to be carried on board a vessel in accordance with the following:

> "(1) The material must be in proper condition for transportation according to the requirements of this subchapter; (2) All packages in the transport vehicle or container must be secured to prevent movement in any direction. Vertical restraint is not required if the shape of the packages, loading pattern, and horizontal restraint preclude vertical movement of the load within the freight container or transport vehicle; (3) Bulkheads made of dunnage which extend to the level of the cargo must be provided unless the packages are stowed flush with the sides or ends; (4) Dunnage must be secured to the floor when the cargo consists of dense materials or heavy packages."

Each freight container must be placarded as required by Subpart F of Part 172 of the hazardous materials transportation regulations [49 CFR 176.76(f)].

Section 49 CFR 176.80 requires that radioactive materials be segregated from other hazardous materials so that they do not interact dangerously in an accident or, alternatively, requires that the radioactive material be in separate holds when stored under deck. In 49 CFR 176.83(b), a table is provided (Table II) that specifies the minimum separation distances for different classes of hazardous materials on board a vessel. A minimum horizontal separation distance of 3 m (10 ft) projected vertically from the reference package is required. For specified hazardous materials, the "separate from" requirement means that the materials must be placed in separate holds when stowed under deck.

Ground Transport

Overland shipments (by rail car or by truck) are regulated by a variety of the Department of Transportation and NRC regulations dealing with packaging, notification, escorts and communication. In addition, there are specific regulations for carriage by truck and carriage by rail.

When provisions are made to secure a package so that its position within the transport vehicle remains fixed during transport, with no loading or unloading between the beginning and end of transport, a package shipped overland in exclusive-use closed transport vehicles may not exceed the following radiation levels as provided in 49 CFR 173.441(b):

1. 200 millirem per hour on the external surface of the package unless the following conditions are met, in which case the limit is 1,000 millirem per hour;

    i. The shipment is made in a closed transport vehicle;

    ii. The package is secured within the vehicle so that its position remains fixed during transportation; and

    iii. There are no loading or unloading operations between the beginning and end of the transportation;

2. 200 millirem per hour at any point on the outer surface of the vehicle, including the top and underside of the vehicle; or in the case of a flat-bed style vehicle, at any point on the vertical planes projected from the outer edges of the vehicle, on the upper surface of the load (or enclosure is used), and on the lower external surface of the vehicle;

3. 10 millirem per hour at any point 2 m (6.6 ft) from the outer lateral surfaces of the vehicle (excluding the top and underside of the vehicle); or in the case of a flat-bed style vehicle, at any point 2 m (6.6 ft) from the vertical planes projected by the outer edges of the vehicle (excluding the top and underside of the vehicle); and

4. 2 millirem per hour in any normally occupied space.

The shipper of record must comply with the requirements of 10 CFR 71.5 and 73.37. Section 71.5 provides that all overland shipments must be in compliance with Department of Transportation and NRC regulations, which provide for security of irradiated reactor fuel. General requirements include:

- Provide notification to NRC in advance of each shipment,
- Develop a shipping plan,
- Provide escort instructions
- Establish a communication center to be staffed 24 hours a day,
- Make arrangements with local law enforcement agencies along the route for their response, if not using law enforcement personnel as escort, ensure that the escorts are trained in accordance with 10 CFR 73.37 Appendix D, and
- Ensure that escorts make notification calls every 2 hours to the communications center.

Additional requirements include having two armed escorts within heavily populated areas (when not in heavily populated areas, only one escort is needed) and the capability of communicating with the communications center and local law enforcement agencies through a radiotelephone or other NRC-approved means of two-way voice communications.

The shipper of record, as required by 49 CFR 173.22 provides

5.5 Emergency Management and Response

5.5.1 Authorities and Directives

Emergency Planning and Community Right-to-Know Act of 1986 (42 USC §11001 et seq.) (also known as "SARA Title III")

Under Subtitle A of this Act, Federal facilities, including those owned by DOE, provide various information (such as inventories of specific chemicals used or stored and releases that occur from these sites) to the State Emergency Response Commission and to the Local Emergency Planning Committee to ensure that emergency plans are sufficient to respond to unplanned releases of hazardous substances. Implementation of the provisions of this Act began voluntarily in 1987, and inventory and annual emissions reporting began in 1988 based on 1987 activities and information. DOE also requires compliance with Title III as a matter of Agency policy. The requirements for this Act were promulgated by the Environmental Protection Agency in 40 CFR Parts 350 through 372.

The Toxic Substances Control Act also regulates the treatment, storage, and disposal of certain toxic substances not regulated by the Resource Conservation and Recovery Act or other statutes, particularly polychlorinated biphenyls, chlorofluorocarbons, and asbestos.

Quantities of Radioactive Materials Requiring Consideration of the Need for an Emergency Plan for Responding to a Release (10 CFR Part 30.72 Schedule C)

This list is the basis for both the public and private sector to determine if the radiological materials they deal with must have an emergency response plan for unscheduled releases. It is one of the threshold criteria documents for DOE Hazards Assessments required by DOE Order 5500.3A, "Planning and Preparedness for Operational Emergencies" (DOE, 1991c).

Occupational Safety and Health Administration Emergency Response, Hazardous Waste Operations and Worker Right to Know (29 CFR)

This regulation sets down the Occupational Safety and Health Administration requirements for employee safety in a variety of working environments. It addresses employee emergency and fire prevention plans (Section 1910.38), hazardous waste operations and emergency response (Section 1910.120), and hazards communication (Section 1910.1200) that enables employees to be aware of the dangers they face from hazardous materials at their workplace.

Emergency Management and Assistance (44 CFR 1.1)

This regulation contains the policies and procedures for the Federal Emergency Management Act, National Flood Insurance Program, Federal Crime Insurance Program, Fire Prevention and Control Program, Disaster Assistance Program, and Preparedness Program including radiological planning and preparedness.

Hazardous Materials Tables & Communications, Emergency Response Information Requirements (49 CFR Part 172)

The regulatory requirements for marking, labeling, placarding, and documenting hazardous materials shipments are defined in this regulation. It also specifies the requirements for providing hazardous material information and training.

Public Law 93-288, as Amended by Public Law 100-707, "Robert T. Stafford Disaster Relief and Emergency Assistance Act," November 23, 1988

The Robert T. Stafford Disaster Relief and Emergency Assistance Act, P.L. 93-288, as amended, provides an orderly and continuing means of assistance by the Federal Government to State and local governments in carrying out their responsibilities to alleviate the suffering and damage resulting from disasters. The President, in response to a State Governor's request, may declare an "emergency" or "major disaster," in order to provide Federal assistance under the Act. The President, in Executive Order 12148, delegated all functions, except those in Sections 301, 401, and 409, to the Director, Federal Emergency Management Agency. The Act provides for the appointment of a Federal Coordinating Officer who will operate in the designated area with a State Coordinating Officer for the purpose of coordinating State and local disaster assistance efforts with those of the Federal Government.

Public Law 96-510, "Comprehensive Environmental Response, Compensation, and Liability Act of 1980," Section 104(i), 42 U.S.C. 9604(i)

More popularly known as "Superfund," this Act provides the needed general authority for Federal and State governments to respond directly to hazardous substances incidents. The Act requires reporting of spills, including radioactive, to the National Response Center.

Public Law 98-473, Justice Assistance Act of 1984

These Department of Justice regulations implement the Emergency Federal Law Enforcement Assistance functions vested in the Attorney General. Those functions were established to assist State and/or local units of government in responding to a law enforcement emergency. The Act defines the term "law enforcement emergency" as an uncommon situation which requires law enforcement, which is or threatens to become of serious or epidemic proportions, and with respect to which State and local resources are inadequate to protect the lives and property of citizens, or to enforce the criminal law. Emergencies that are not of an ongoing or chronic nature, such as the Mount Saint Helens volcanic eruption, are eligible for Federal law enforcement assistance. Such assistance is defined as funds, equipment, training, intelligence information, and personnel. Requests for assistance must be submitted in writing to the Attorney General by the chief executive office of a State. The Plan does not cover the provision of law enforcement assistance. Such assistance will be provided in accordance with the regulations referred to in this paragraph [28 CFR Part 65, implementing the Justice Assistance Act of 1984] or pursuant to any other applicable authority of the Department of Justice.

Communications Act of 1934, as Amended

This Act gives the Federal Communications Commission emergency authority to grant Special Temporary Authority on an expedited basis to operate radio frequency devices.

5.5.2 Executive Orders

Executive Order 10480, as Amended, "Further Providing for the Administration of the Defense Mobilization Program," August 1953

Part II of the Order delegates to the Director, Federal Emergency Management Agency, with authority to redelegate, the priorities and allocation functions conferred on the President by Title I of the Defense Production Act of 1950, as amended.

Executive Order 12148, "Federal Emergency Management," July 20, 1979

Executive Order 12148 transferred functions and responsibilities associated with Federal emergency management to the Director, Federal Emergency Management Agency. The Order assigns the Director, Federal Emergency Management Agency, the responsibility to establish Federal policies for and to coordinate all civil defense and civil emergency planning, management, mitigation, and assistance functions of Executive Agencies.

Executive Order 12472, "Assignment of National Security and Emergency Preparedness Telecommunications Functions," April 3, 1984

Executive Order 12472 establishes the National Communication System. The National Communication System consists of the telecommunications assets of the entities represented on the National Communication System Committee of Principals and an administrative structure consisting of the Executive Agent, the National Communication System Committee of Principals, and the Manager. The National Communication System Committee of Principals consists of representatives from those Federal departments, agencies, or entities, designated by the President, which lease or own telecommunications facilities or services of significance to national security or emergency preparedness.

Executive Order 12656, "Assignment of Emergency Preparedness Responsibilities," November, 1988

This order assigns emergency preparedness responsibilities to Federal departments and agencies.

5.5.3 Emergency Planning Documents

"Federal Radiological Emergency Response Plan," November 1985

This document is to be used by Federal agencies in peacetime radiological emergencies. It primarily concerns the off-site Federal response in support of State and local governments with jurisdiction for the emergency. The Federal Radiological Emergency Response Plan provides the Federal Government's concept of operations based on specific authorities for responding to radiological emergencies, outlines Federal policies and planning assumptions that underlie this concept of operations and on which Federal agency response plans were based, and specifies authorities and responsibilities of each Federal agency that may have a significant role in such emergencies.

"National Plan for Telecommunications Support [in Non-Wartime Emergencies," January 1992

This plan provides guidance in planning for and providing telecommunications support for Federal agencies involved in emergencies, major disasters, and other urgent events, excluding war.

Department of Defense Directive 3025.1, "Military Support to Civil Authorities," 1992

This directive outlines Department of Defense policy on assistance to the civilian sector during disasters and other emergencies. Use of the Department of Defense military resources in civil emergency relief operations will be limited to those resources not immediately required for the execution of the primary defense mission. Normally, the Department of Defense military resources will be committed as a supplement to non-Department of Defense resources that are required to cope with the humanitarian and property protection requirement caused by the emergency. In any emergency, commanders are authorized to employ Department of Defense resources to save lives, prevent human suffering, or mitigate great property loss. Upon declaration of a major disaster under the provisions of P.L. 93-288, as amended, the Secretary of the Army is the Department of Defense Executive Agent, and the Director of Military Support is the action agent for civil emergency relief operations. Military personnel will be under command of and directly responsible to their military superiors and will not be used to enforce or execute civil law in violation of 18 U.S.C. 1385, except as otherwise authorized by law. Military resources shall not be procured, stockpiled, or developed solely to provide assistance to civil authorities during emergencies.

Federal Preparedness Circular 8, "Public Affairs in Emergencies"

This Circular establishes the Interagency Committee on Public Affairs in Emergencies to coordinate public information planning and operations for management of emergency information. The Circular was reviewed in draft by the Interagency Committee on Public Affairs in Emergencies and will receive formal department and agency review.

American Red Cross Disaster Services Regulations and Procedures, ARC 3003, January 1984

This document details the delegation of disaster services program responsibilities to officials and units of the American Red Cross. Also defined are the American Red Cross administrative regulations and procedures for disaster planning, preparedness, and response.

Statement of Understanding between the Federal Emergency Management Agency and the American National Red Cross, January 22, 1982

The statement of understanding between the Federal Emergency Management Act and the American National Red Cross describes major responsibilities in disaster preparedness planning and operations in the event of a war-caused national emergency or a peacetime disaster, outlines areas of mutual support and cooperation, and provides a frame of reference for similar cooperative agreements between State and local governments and the operations headquarters and chapters of the American Red Cross.

6. List of Preparers

Name:	Charles R. Head
Affiliation:	Department of Energy
Education:	MS, Control Theory, George Washington University BS, Electrical Engineering, Rice University
Experience/ Technical Specialty:	Twenty-seven years. Nuclear safety oversight, safeguards and security requirements, Tiger Team assessments, spent fuel storage.
EIS Responsibility:	DOE Foreign Research Reactor Spent Nuclear Fuel Project Manager

Name:	Patrick J. Wells
Affiliation:	Department of Energy
Education:	MS, Engineering Management, George Washington University BS, Civil Engineering, Marquette University
Experience/ Technical Specialty:	Eleven years. Occupational safety and health, oversight system acquisition, reliability, and maintainability.
EIS Responsibility:	Assistant to the DOE Project Manager

Name:	Darren W. Piccirillo
Affiliation:	Department of Energy
Education:	BS, Physics, Southern Connecticut State University
Experience/ Technical Specialty:	Eight years. Reactor plant construction and operation.
EIS Responsibility:	Spent fuel characterization, storage impact analysis

Name:	Eleanor R. Busick
Affiliation:	Department of State
Education:	MA, Economics, Yale University BA, Oberlin College
Experience/ Technical Specialty:	Nineteen years. Nuclear nonproliferation policy, economics (fuel markets, finance).
EIS Responsibility:	Nonproliferation and national policies

Name:	Fred McGoldrick
Affiliation:	Department of State
Education:	PhD, Political Science, American University BA, Boston College
Experience/ Technical Specialty:	Twenty-three years. Nuclear nonproliferation policy.
EIS Responsibility:	Nonproliferation and national policies

Name:	Ibrahim H. Zeitoun
Affiliation:	Science Applications International Corporation
Education:	PhD, Environmental Sciences, Michigan State University MS, Fisheries, Michigan State University BS, Chemistry & Zoology, University of Alexandria
Experience/ Technical Specialty:	Twenty-two years. Waste management, environmental assessment, and NEPA compliance.
EIS Responsibility:	Overall EIS Project Manager. NEPA compliance, spent fuel management environmental justice

Name:	Audrey J. Adamson
Affiliation:	Science Applications International Corporation
Education:	BA, University of Michigan MPA, George Washington University
Experience/ Technical Specialty:	Eleven years. International, environmental, and technology policy analysis, public outreach, and technical writing.
EIS Responsibility:	Emergency response and communication planning

Name:	Douglas H. Amick
Affiliation:	Science Applications International Corporation
Education:	AS, Law Enforcement, Walter State Community College
Experience/ Technical Specialty:	Twenty years. Marine ports, port safety inspections and security, plant operations, transportation risks.
EIS Responsibility:	Port safety, operation and characteristics

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Name: Paula W. Austin

Affiliation: Science Applications International Corporation

Education: BS, Management and Technology, University of Maryland

Experience/ Technical Specialty: Nineteen years. Nuclear and environmental policy analysis, public outreach, technical writing.

EIS Responsibility: EIS Summary, public hearings task leader

Name: Rakesh Bahadur

Affiliation: Science Applications International Corporation

Education: PhD, Groundwater Hydrology, Colorado State University
MS, Groundwater Hydrology, Colorado State University
MSc, Geology, Punjab University

Experience/ Technical Specialty: Fifteen years. Hydrology, site characterization, environmental assessment, and risk assessment.

EIS Responsibility: Affected environment, Geographic Information Systems, Environmental Justice

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Name: Bruce M. Biwer

Affiliation: Argonne National Laboratory

Education: PhD, Chemistry, Princeton University
MA, Chemistry, Princeton University
BA, Chemistry, St. Anselm College

Experience/ Technical Specialty: Fourteen years. Radiological pathway analysis, dose calculations, radiological transportation risk analysis.

EIS Responsibility: Radiological transportation risk analysis

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Name: Keith R. Brown

Affiliation: Science Applications International Corporation

Education: MS, Mathematics, Idaho Sate University
BS, Mathematics, Idaho State University

Experience/ Technical Specialty: Thirty-four years. Transportation casks, research reactor operations, and

Name:	Burrus M. Carnahan
Affiliation:	Science Applications International Corporation
Education:	LLM, International Law, University of Michigan JD, Northwestern University BA, Political Science, Drake University
Experience/ Technical Specialty:	Twenty-one years. Weapons proliferation, international agreements, and arms control.
EIS Responsibility:	Nonproliferation policies

Name:	Harry Chernoff
Affiliation:	Science Applications International Corporation
Education:	MBA, Marymount University BA, Economics, College of William & Mary
Experience/ Technical Specialty:	Seventeen years. Economics, socioeconomics, finance, and engineering economics.
EIS Responsibility:	Costs, and Socioeconomics

Name:	Louis Cofone, Jr.
Affiliation:	Science Application International Corporation
Education:	MS, Microbiology, University of Dayton BS, Biology, Philadelphia College of Pharmacy and Science
Experience/ Technical Specialty:	Twenty years. Computer information analysis, data base development and management, environmental and health risk assessment.
EIS Responsibility:	Development and management of the comment tracking and document control system

Name:	Cecil C. Cross, III
Affiliation:	Science Applications International Corporation
Education:	MEM, Environmental Management, Duke University BA, Biology, Gettysburg College
Experience/ Technical Specialty:	Thirteen years. Regulatory compliance, environmental assessment, and hazardous and mixed waste management.
EIS Responsibility:	Port operation, port environmental and climatic conditions

Name:	Larry Danese
Affiliation:	Science Applications International Corporation
Education:	MBA, Florida International University BS, Electrical Engineering, University of Florida
Experience/ Technical Specialty:	Twenty-three years. Cask design, transportation systems, emergency response, and regulatory compliance.
EIS Responsibility:	Transportation casks and regulations, port operation regulations and activities

Name:	Gary M. DeMoss
Affiliation:	Science Applications International Corporation
Education:	MS, Engineering Administration, Virginia Polytechnic Institute BS, Mechanical Engineering, University of Virginia
Experience/ Technical Specialty:	Thirteen years. Risk analysis, reliability and safety engineering, uranium enrichment, and transportation.
EIS Responsibility:	Ground transportation safety and impact analysis, port selection and operation

Name:	Scott E. Drummond, Jr.
Affiliation:	Science Applications International Corporation
Education:	BS, Marine Transportation, SUNY Maritime College
Experience/ Technical Specialty:	Forty-two years. Strategic sealift, logistics support, ocean survey, nautical charting, SWATH ship design and operation.
EIS Responsibility:	Port information

Name:	Habib A. Durrani
Affiliation:	Science Applications International Corporation
Education:	BSc, Engineering Science, Peshawar University
Experience/ Technical Specialty:	Twenty years. Nuclear facilities operation, design maintenance regulations and safety
EIS Responsibility:	Chemical separation technologies

Name:	Barbara M. Ebert
Affiliation:	Science Applications International Corporation
Education:	MA, National Security Studies, Georgetown University BS, Foreign Service, Comparative and Regional Studies, Georgetown University
Experience/ Technical Specialty:	Twelve years. Weapons proliferation.
EIS Responsibility:	Nonproliferation policies

Name:	Martin W. Ebert
Affiliation:	Science Applications International Corporation
Education:	BSc, Nuclear Engineering, University of Arizona MSc, Applied Physics, University of Strathclyde
Experience/ Technical Specialty:	Twenty-four years. Nuclear powerplant operations, spent fuel technology, and technical safety requirements.
EIS Responsibility:	Marine and port safety and impact analysis, public hearings response coordinator

Name:	Daniel W. Gallagher
Affiliation:	Science Applications International Corporation
Education:	MS, Nuclear Engineering, Rensselaer Polytechnic Institute BS, Nuclear Engineering, Rensselaer Polytechnic Institute
Experience/ Technical Specialty:	Fifteen years. Reliability and risk engineering, probabilistic safety assessment, plant design, and regulatory analysis.
EIS Responsibility:	Marine and port safety and impact analysis

Name:	Reginald L. Gotchy
Affiliation:	Science Applications International Corporation
Education:	PhD, Radiation Biology, Colorado State University MS, Radiation Health, Colorado State University BS, Zoology, University of Washington
Experience/ Technical Specialty:	Twenty-six years. NEPA compliance, safety analysis, risk assessment, radiation biology, health physics (Certified Health Physicist), and emergency response planning.
EIS Responsibility:	Port selection and radiological consequences and health effects

Name:	Peter Grier
Affiliation:	Science Applications International Corporation
Education:	BS, Psychology, University of Maryland
Experience/ Technical Specialty:	Twenty years. Emergency management, commercial nuclear energy, quality assurance, and transportation regulatory compliance.
EIS Responsibility:	Emergency response, security, and communication planning

Name:	Timothy T. Holmes
Affiliation:	Science Applications International Corporation
Education:	JD, University of Kansas School of Law BA, Business, Washburn University
Experience/ Technical Specialty:	Four years. Legal and environmental analysis, involving NEPA, RCRA, and other environmental regulations, document review and contract compliance.
EIS Responsibility:	Comment Response Document task leader, technical editor

Name:	Joseph W. James
Affiliation:	Science Applications International Corporation
Education:	BSG, Administration of Justice, American University MA, Management, Central Michigan University PhD, Environmental Science, LaSalle University
Experience/ Technical Specialty:	Thirty years. Nuclear safeguards and security, standards development, quality assurance, regulatory analysis, and licensing support.
EIS Responsibility:	Safeguards and security planning

Name:	Roy Karimi
Affiliation:	Science Applications International Corporation
Education:	ScD, Nuclear Engineering, Massachusetts Institute of Technology NE, Nuclear Engineering, Massachusetts Institute of Technology MS, Nuclear Engineering, Massachusetts Institute of Technology BSc, Chemical Engineering, Abadan Institute of Technology
Experience/ Technical Specialty:	Fourteen years. Nuclear powerplant safety, risk and reliability analysis, design analysis, and probabilistic risk assessment.
EIS Responsibility:	Spent fuel characterization, accident and impact analysis, quality control reviews

Name:	Stephen J. Krill, Jr.
Affiliation:	Science Applications International Corporation
Education:	BS, Nuclear and Power Engineering, University of Cincinnati
Experience/ Technical Specialty:	Five years. Safety and risk analysis, reactor and fuel processing system design, operation and inspection, and emergency preparedness.
EIS Responsibility:	Transportation cask descriptions, environmental consequences

Name:	Merritt E. Langston, PE
Affiliation:	Science Applications International Corporation
Education:	BS, MS, Metallurgical Engineering, Missouri School of Mines
Experience/ Technical Specialty:	Thirty-two years. Quality management, nuclear engineering, defense programs, nuclear waste management. Six years. Reactor containment materials development.
EIS Responsibility:	Technical reviews, quality control task leader

Name:	Christi D. Leigh
Affiliation:	Sandia National Laboratories
Education:	PhD, Engineering, University of New Mexico MS, Chemical Engineering, Stanford University BS, Chemical Engineering, Arizona State University
Experience/ Technical Specialty:	Six years. Radioactive and hazardous waste management and minimization. Five years. Nuclear reactor safety.
EIS Responsibility:	At-sea submerged cask risk assessment

Name:	Charles D. Massey
Affiliation:	Sandia National Laboratories
Education:	PhD, Radiation Health, University of Pittsburgh MS, Health Physics, University of Pittsburgh BS, Marine Transportation, U.S. Merchant Marine Academy
Experience/ Technical Specialty:	Thirteen years. NEPA, risk assessment, transportation and energy technology evaluation.
EIS Responsibility:	Marine transportation risk assessment and impacts

Name:	Ronya J. McMillen
Affiliation:	Science Applications International Corporation
Education:	MA, International Science & Technology Policy, George Washington University BS, Sociology, Chatham College
Experience/ Technical Specialty:	Thirteen years. Technical analysis and report writing. Four years public outreach and policy analysis on domestic and foreign nuclear technology regulatory issues.
EIS Responsibility:	EIS Summary Coordinator, Public comment and hearing summaries

Name: Charles D. Miller

Affiliation: Science Applications Internal Corporation

Education: BS, Marine Transportation, SUNY Maritime College

Experience/ Technical Specialty: Forty years. Transportation/distribution system design, planning, implementation,.and management.

EIS Responsibilty: Port information

Name: Todd Miller

Affiliation: Science Applications International Corporation

Education: BS, Civil Engineering, Worcester Polytechnic Institute

Experience/ Technical Specialty: Four years. Safety analysis, environmental assessment, NEPA compliance.

EIS Responsibility: Accident analysis, radiological consequences

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Name: Steven M. Mirsky

Affiliation: Science Applications International Corporation

Education: MS, Nuclear Engineering, Pennsylvania State University
BS, Mechanical Engineering, The Cooper Union

Experience/ Technical Specialty: Nineteen years. Safety analysis, nuclear powerplant design, operations, and foreign nuclear powerplant system analysis.

EIS Responsibility: Storage technology, safety and impact analysis

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Name: Frederick A. Monette

Affiliation: Argonne National Laboratory

Education: MS, Health Physics, Colorado State University
BA, Physics, St. Johns University

Experience/ Technical Specialty: Six years. Radiological risk assessment, radiological transportation risk analysis, dose calculations.

EIS Responsibility: Radiological transportation risk and impacts analysis

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Name: Michael Moore

Affiliation: Science Applications International Corporation

Education: BA, Economics, University of Maryland

Experience/ Technical Specialty: Twelve years. Analysis and design of environmental/waste information systems, drafting and editing of technical documents for energy, environmental, and defense initiatives.

EIS Responsibility: Quality control reviews, technical editor

Name:	Alexander P. Murray
Affiliation:	Science Applications International Corporation
Education:	MS, Chemical Engineering, Carnegie-Mellon University BS, Chemical Engineering, Carnegie-Mellon University

Name:	Van Romero
Affiliation:	Sandia National Laboratories
Education:	PhD, Physics, State University of New York MS, Physics, New Mexico Tech BS, Physics, New Mexico Tech
Experience/ Technical Specialty:	Fifteen years. Environmental health physics and radiation protection, NEPA compliance, DOE order compliance, environmental impact testing, risk assessment, nuclear safety, health physics, radiation transport, and nuclear emergency response.
EIS Responsibility:	Radiation exposure analysis for marine transport

Name:	William B. Samuels
Affiliation:	Science Applications International Corporation
Education:	PhD, Biology, Fordham University MS, Marine Science, Long Island University BS, Biology & Geology, University of Rochester
Experience/ Technical Specialty:	Sixteen years. Geographic Information Systems, computer simulation and mathematical modeling, environmental database management systems.
EIS Responsibility:	Geographic Information Systems, environmental justice

Name:	Elizabeth C. Saris
Affiliation:	Science Applications International Corporation
Education:	BA, Political Science, George Washington University
Experience/ Technical Specialty:	Fifteen years. Energy and environmental policy analysis, public outreach, and technical writing.
EIS Responsibility:	EIS Summary, public hearings support

Name:	Patrick R. Schwab
Affiliation:	Science Applications International Corporation
Education:	PhD, Nuclear Engineering, University of Wisconsin MS, Nuclear Engineering, University of Wisconsin BS, Nuclear Engineering, Kansas State University
Experience/ Technical Specialty:	Eighteen years. Design criteria, technical safety surveys, foreign nuclear technology analysis, configuration studies, and spent fuel reprocessing.
EIS Responsibility:	Environmental and policy consequences, chemical separation technologies and impacts

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Name: Barry Smith

Affiliation: Science Applications International Corporation

Education: JD, George Washington University National Law Center
BA, Political Science, Indiana University

Experience/ Technical Specialty: Twenty-three years. NEPA compliance, environmental law, regulatory compliance, and waste management.

EIS Responsibility: Environmental regulation/compliance

Name: Jeremy L. Sprung

Affiliation: Sandia National Laboratories

Education: PhD, Physical-Organic Chemistry, UCLA
BA, Chemistry, Yale University

Experience/ Technical Specialty: Twenty-nine years. Photochemistry and air pollution, reactor accident consequences, reactor safety studies, and transportation risk assessment.

EIS Responsibility: Port accident risk analysis

Name: Donna J. Stucky

Affiliation: Pacific Northwest Laboratories

Education: MS, Agricultural Economics, Purdue University
BA, Economics, Pacific Lutheran University

Experience/ Technical Specialty: Two years. Economic research.

EIS Responsibility: Environmental consequences

Name: Robert Wayland

Affiliation: Science Applications International Corporation

Education: PhD, Atmospheric Science, North Carolina State University
MS, Environmental Science, University of Virginia
BA, Environmental Science, University of Virginia

Experience/ Technical Specialty: Eleven years. Boundary-layer meteorology, atmospheric structure and composition, ocean-atmosphere interactions, atmospheric modeling.

EIS Responsibility: Port meteorological data assessments, site nonradiological impact analyses

Name:	Timothy Wheeler
Affiliation:	Sandia National Laboratories
Education:	MS, Systems Engineering, University of Virginia BS, Mechanical Engineering, University of New Hampshire
Experience/ Technical Specialty:	Fourteen years. NEPA compliance, radioactive material transportation risk analysis, probabilistic risk assessment.
EIS Responsibility:	At-sea submerged cask risk assessment

Name:	John W. Williams
Affiliation:	Science Applications International Corporation
Education:	PhD, Physics, New Mexico State University MS, Physics, New Mexico State University BS, Mathematics, North Texas State University
Experience/ Technical Specialty:	Twenty years. NEPA compliance, electromagnetic models, air quality modeling, ionizing radiation impacts and safety.
EIS Responsibility:	Environmental justice, ports selection, quality control reviews

Name:	Steven E. Wujciak
Affiliation:	U.S. Department of Transportation, Research & Special Projects Administration, Volpe National Transportation System Center
Education:	MBA, Anna Maria College BS, Business Administration, Anna Maria College
Experience/ Technical Specialty:	Fifteen years. Operations research, transportation analysis, emergency preparedness.
EIS Responsibility:	Ground transportation analysis

Name:	Maron D. Wylie
Affiliation:	U.S. Department of Transportation, Research & Special Projects Administration, Volpe National Transportation System Center
Education:	MS, Math and Computer Science, Worcester State College BS, Business Administration, University of Southern Mississippi
Experience/ Technical Specialty:	Fifteen years. Operations research, transportation analysis, emergency preparedness.
EIS Responsibility:	Ground transportation analysis

Name	Michael R. Zanotti
Affiliation:	Science Applications International Corporation
Education:	MPA, Administrative Management and Organization, Golden Gate University MPA, Health Services Administration, Golden Gate University BA, Behavioral Sciences, University of Maine AA, Criminal Justice, University of Maine
Experience/ Technical Specialty:	Fifteen years. Certified Emergency Manager (CEM), Emergency management, emergency response, fire response, hazardous materials response, facilities operation.
EIS Responsibility:	Emergency management and response

7. Agencies Consulted

The following agencies were consulted in the development of this Draft Environmental Impact Statement.

Federal Agencies

Arms Control and Disarmament Agency
Military Traffic Management Command
Military Ocean Terminal, Oakland (CA)
Military Ocean Terminal, Sunny Point (NC)
Naval Weapons Station, Concord (CA)
Naval Weapons Station, Charleston (SC)
Port Hueneme (CA) Naval Construction Battalion Center
U.S. Department of Defense
U.S. Department of Army
U.S. Coast Guard
U.S. Merchant Marine Academy
U.S. Fish and Wildlife Service

State Agencies

Alabama Department of Conservation
  and Natural Resources
Alabama Department of Environmental Management,
  Water Quality Division
Alabama Natural Heritage Program
Alabama State Docks, Mobile (AL)
California Fish & Game Heritage Program
California Regional Water Quality Control Board,
  San Francisco Bay Region
Delaware Department of Natural Resources and
  Environmental Control, Division of Water Resources
Delaware Natural Heritage Inventory
Florida Department of Environmental Regulation,
  Bureau of Surface Water Management
Florida Natural Areas Inventory
Fort Clinch State Park, Amelia Island, FL
Georgia Department of Natural Resources,
  Environmental Protection Division
Mississippi Department of Environmental Quality,
  Water Quality Division
Mississippi Natural Heritage Program
Mississippi State Port Authority at Gulfport
New Hampshire Port Authority
New Jersey Natural Heritage Program
North Carolina Department of Environment, Health,
  and Natural Resources, Division of Environmental
  Management
North Carolina Natural Heritage Program
North Carolina State Ports Authority
Oregon Natural Heritage Program
Pennsylvania Department of Environmental Resources,
  Water Quality Division
Pennsylvania Natural Diversity Inventory
Ports Authority of New York & New Jersey
South Carolina Department of Health and
  Environmental Control, Water Quality Division

Local Agencies (Continued)

Port of Fernandina (FL)
Port of Galveston (TX)
Port of Grays Harbor (WA)
Port of Houston Authority (TX)
Port of Hueneme (CA)
Port of Long Beach (CA)
Port of Longview (WA)
Port of Los Angeles (CA)
Port of Miami (FL)
Port of New Haven (CT)
Port of Oakland (CA)
Port of Palm Beach (FL)
Port of Port Arthur (TX)
Port of Portland (ME)
Port of Portland (OR)
Port of Portsmouth (NH)
Port of Richmond (CA)
Port of Richmond Commission (VA)
Port of San Francisco (CA)
Port of Seattle (WA)
Port of Tacoma (WA)
Port of Vancouver, U.S.A. (WA)
Port of Wilmington (DE)
Port of Wilmington (NC)
San Diego Unified Port District (CA)
Tampa Port Authority (FL)

Other

Australian Nuclear Science & Technology Organization (ANSTO)
Austrian Research Centre, Austria
Belgian Nuclear Research Centre
GKSS Research Center, Germany
Hahn-Meitner Institut Berlin, Germany
Interfaculty Reactor Institute, Delft University of Technology, The Netherlands
Joint Research Centre-Petten, Institute for Advanced Materials, The Netherlands
National Center for Scientific Research, "Demokritos," Greece
Paul Scherrer Institute, Switzerland
RISO National Laboratory, Denmark
Studsvik Nuclear AB, Sweden
United Kingdom Atomic Energy Authority, Thurso, Dounreay Caithness, Scotland

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9. Glossary

Absorbed dose. The energy imparted by ionizing radiation per unit mass of irradiated material. The unit of absorbed dose is the rad.

Accident. An unplanned sequence of events that results in undesirable consequences.

Actinide. Any of a series of chemically similar, mostly synthetic, radioactive elements with atomic numbers ranging from actinium (89) through lawrencium (103).

Acute exposure. A single exposure to a toxic substance which may result in severe biological harm or death. Acute exposures are usually characterized as lasting no longer than a day.

Alpha-emitter. A radioactive substance that decays by releasing an alpha particle.

Alpha particle. A particle consisting of two protons and two neutrons, given off by the decay of many elements, including uranium, plutonium, and radon. Alpha particles cannot penetrate a sheet of paper. However, alpha emitting isotopes in the body can be very damaging.

As low as reasonably achievable (ALARA). The approach to radiation protection to manage and control exposures (both individual and collective) to the work force and to the general public to as low as is reasonable, taking into account social, technical, economic, practical, and public policy considerations. ALARA is not a dose limit but a process which has the objective of attaining doses as far below the applicable limits as is reasonably achievable.

Atomic Energy Act (AEA). A law passed in 1954 that placed nuclear production and control of nuclear materials within a civilian agency, originally the Atomic Energy Commission. The Atomic Energy Commission was replaced by the U.S. Nuclear Regulatory Commission, the U.S. Department of Energy, and predecessor agencies (i.e., ERDA, FERC).

Atomic number. The number of positively charged protons in the nucleus of an atom or the number of electrons on an electrically neutral atom.

Background radiation. Radiation from: (1) Naturally occurring radioactive materials which have not been technologically enhanced, (2) cosmic sources, (3) global fallout as it exists in the environment (such as from the testing of nuclear explosive devices), (4) radon and its progeny in concentrations or levels existing in buildings or the environment which have not been elevated as a result of current or prior activities, and (5) consumer products containing nominal amounts of radioactive material or producing nominal amounts of radiation.

Beta particle. A particle emitted in the radioactive decay of many radionuclides. A beta particle is identical with an electron. It has a short range in air and a low ability to penetrate other materials.

Canning. The process of placing spent nuclear fuel in canisters to retard corrosion, contain radioactive releases, or control geometry.

Cask. A heavily shielded massive container for holding nuclear materials during shipment.

Characterization. The determination of waste or spent nuclear fuel composition and properties, whether by review of process knowledge, nondestructive examination or assay, or sampling and analysis, generally done to determine appropriate storage, treatment, handling, transportation, and disposal requirements.

Chemical separation. A process for extracting uranium and plutonium from dissolved spent nuclear fuel and irradiated targets. The fission products that are left behind are high level wastes. Chemical separation is also known as reprocessing.

Cladding. The outer layer of metal over the fissile material of a nuclear fuel element. Cladding on the Department of Energy's spent fuel is usually aluminum, zirconium, or stainless steel.

Collective dose. The sum of the total effective dose equivalents of all individuals in a specified population. Collective dose is expressed in units of person-rem (or person-sievert).

Committed effective dose equivalent. The sum of the committed dose equivalents to various tissues in the body, each multiplied by the appropriate weighting factor. Committed effective dose equivalent is expressed in units of rem (or sievert), and will be accumulated during the fifty years following an intake of radioactive material into an individual's body.

Competitive fee. A fee that could be charged to foreign research reactor operators related to the estimated cost of spent nuclear fuel management and disposal outside the United States.

Conditioning. See stabilization (of spent nuclear fuel).

Contact-handled waste. Packaged waste whose external surface dose rate does not exceed 200 [illegible] hour.

Contamination. The deposition of undesirable radioactive material on the surfaces of structures, areas, objects, or personnel.

Core. The central portion of a nuclear reactor containing the fuel elements, moderator, neutron poisons,

Degraded (spent nuclear fuel). See failed fuel.

Depleted uranium. Uranium that, through the process of enrichment, has been stripped of most of the uranium-235 it once contained, so that it has more uranium-238 than natural uranium. It is used as shielding, in some parts of nuclear weapons, and as a raw material for plutonium production.

Developed countries. Countries with high-income economies (World Bank, 1994).

Developing countries. Countries with other-than-high-income economies (World Bank, 1994).

Discounted dollars. Expressing income and expenditures that occur over time as if they occurred at a common point in time.

Disposal of fuel. Emplacement of fuel to ensure its isolation from the biosphere, with no intention of retrieval.

DOE Orders. Requirements internal to the U.S. Department of Energy (DOE) that establish DOE policy and procedures, including those for compliance with applicable laws.

Dose (or radiation dose). A generic term that means absorbed dose, dose equivalent, effective dose equivalent, committed effective dose equivalent, or total effective dose equivalent as defined elsewhere in this glossary.

Dose rate. The radiation dose delivered per unit time (e.g., rem per year).

Dry storage. Storage of spent nuclear fuel in environments where the fuel is not immersed in water for purposes of both cooling and shielding.

Ecology. The relationship of living things to one another and their environment, or the study of such relationships.

Effective dose equivalent. The summation of the products of the dose equivalent received by specified tissues of the body and the appropriate weighting factor. It includes the dose from radiation sources internal and/or external to the body. The effective dose equivalent is expressed in units of rem (or sievert).

Endangered species. Animals, birds, fish, plants, or other living organisms threatened with extinction by man-made or natural changes in their environment. Requirements for declaring a species endangered are contained in the Endangered Species Act.

Enriched uranium. Uranium that has greater amounts of the isotope uranium-235 than occurs naturally. Naturally occurring uranium is 0.72 percent uranium-235.

Environmental monitoring. The process of sampling and analysis of environmental media in and around a facility being monitored for the purpose of (1) confirming compliance with performance objectives and (2) early detection of any contamination entering the environment to facilitate timely remedial action.

Escalation. A real change in the price level of a particular good or service, unrelated to inflation.

Existing facilities. Facilities that existed at an active DOE site as of the Record of Decision for this Environmental Impact Statement.

Failed fuel. Spent nuclear fuel whose external cladding has cracked, pitted, corroded, or potentially allows the leakage of radioactive gases.

Fissile material. Any material fissionable by thermal (slow) neutrons; the two primary fissile isotopes are uranium-235 and plutonium-239.

Fission. The splitting or breaking of a nucleus into at least two other nuclei and the release of a relatively large amount of energy. Two or three neutrons are usually released during this type of transformation.

Fission products. The nuclei produced by fission of heavy elements, and their radioactive decay products.

Fissionable material. Commonly used as a synonym for fissile material, the meaning of this term has been extended to include material that can be fissioned by fast neutrons, such as uranium-238.

Fuel elements. Nuclear reactor fuel including both the fissile and the structural material serves as cladding.

Full-cost recovery fee. A fee that could be charged to foreign research reactor operators that recovers all costs incurred by the United States for management of their spent nuclear fuel.

Gamma ray. Very penetrating electromagnetic radiation of nuclear origin. Except for origin and energy level, identical to x-rays. Electromagnetic radiation frequently accompanying alpha and beta emissions as radioactive materials decay.

Geologic repository. A place to dispose of radioactive waste deep beneath the earth's surface.

Groundshine. The radiation dose received from radioactive material deposited on the ground's surface.

Half-life. The time in which one-half of the atoms of a particular radioactive substance disintegrate to another nuclear form.

Hazardous material. A substance or material in a quantity and form which may pose an unreasonable risk to health and safety or property when transported in commerce.

Hazardous substance. Any substance that when released to the environment in an uncontrolled or unpermitted fashion becomes subject to the reporting and possible response provisions of the Clean Water Act and the Comprehensive Environmental Response, Compensation, and Liability Act.

Hazardous waste. (1) Wastes that are identified or listed in 40 CFR 261.31 and 261.32. Source, special nuclear material, and by-product material as defined by the Atomic Energy Act of 1954, as amended, are specifically excluded from the term hazardous wastes. (2) As defined in RCRA, a solid waste, or combination of wastes, that because of its quantity, concentration, or physical, chemical, or infectious characteristics, may cause or significantly contribute to an increase in mortality or serious, irreversible, or incapacitating reversible illness or pose a substantial present or potential hazard to human health or the environment when improperly treated, stored, transported, or disposed of, or otherwise managed. (3) By-products of society that can pose a substantial or potential hazard to human health or the environment when improperly managed. Possesses at least one of four characteristics (ignitability, corrosivity, reactivity, or toxicity).

High-efficiency particulate air (HEPA) filter. A filter with an efficiency of at least 99.95 percent used to remove particles from air exhaust streams prior to releasing to the atmosphere.

High-level waste. The highly radioactive waste material that results from the reprocessing of spent nuclear fuel, including liquid waste produced directly from reprocessing and any solid waste derived from the liquid that contains a combination of transuranic and fission product nuclides in quantities that require permanent isolation. High-level waste may include the highly radioactive material that the NRC, consistent with existing law, determines by rule requires permanent isolation.

Inflation. A change in the nominal price level of all goods or services, unrelated to the real escalation of a particular good or service.

Isotopes. Different forms of the same chemical element that differ only by the number of neutrons in their nucleus. Most elements have more than one naturally occurring isotope. Many more isotopes have been produced in reactors and scientific laboratories.

Latent cancer fatalities (LCF). Deaths occurring at later years from radiation-induced cancers.

Levelization. Conversion of a stream of values that vary at a uniform rate over time to a constant value over the same period of time.

Life cycle costs. All costs except the cost of personnel occupying the facility incurred from the time that space requirement is defined until the facility passes out of the government's hands.

Low enriched uranium (LEU). Uranium enriched until it consists of up to 20 percent uranium-235. Used as nuclear reactor fuel.

Low-level waste. A catchall term for any radioactive waste that is not spent fuel, high-level, or transuranic waste.

Management (spent nuclear fuel). Emplacing, operating, and administering facilities, transportation systems, and procedures in order to ensure safe and environmentally responsible handling and storage of spent nuclear fuel pending (and in anticipation of a decision on ultimate disposition. Spent nuclear fuel management also includes activities such as stabilization, examination/characterization, processing or chemical separation, and research and development; including activities that may be necessary to prepare spent nuclear fuel for ultimate disposition.

Maximally exposed individual (MEI). A theoretical individual living at the site boundary receiving the maximum exposure. The individual is assumed to be located in a direction downwind from the release point.

Maximally exposed worker. A marine transport worker, port worker, ground transport worker, or onsite radiation worker who could receive the maximum radiation exposure in a given situation.

Maximum contaminant level (MCL). The maximum permissible levels of a contaminant in water which is delivered to the free flowing outlet of the ultimate user of a public water system, except in the case of turbidity where the maximum permissible level is measured at the point of entry to the distribution system. Contaminants added to the water under the circumstances controlled by the user, except those resulting from corrosion of piping and plumbing caused by water quality, are excluded from this definition.

Metric tons of heavy metal (MTHM). Quantities of unirradiated and spent nuclear fuel and targets are traditionally expressed in terms of metric tons of heavy metal (typically uranium), without the inclusion of other materials, such as cladding, alloy materials, and structural materials. A metric ton is 1,000 kilograms, which is equal to about 2,200 pounds.

National Environmental Policy Act. A Federal law, enacted in 1970, that requires the Federal government to consider the environmental impacts of, and alternatives to, major proposed actions in its decisionmaking processes. Commonly referred to by its acronym, NEPA.

Natural phenomena accidents. Accidents that are initiated by phenomena such as earthquakes, tornadoes, floods, etc.

Nearest public access individual (NPAI). A theoretical individual located at the point of nearest public access to a DOE facility, usually during an accident situation.

Net present value. The value of a series of future income and expense streams brought forward to the present at the discount rate.

Neutron. Uncharged elementary particles with a mass slightly greater than that of the proton, and found in the nucleus of every atom heavier than hydrogen.

Nonproliferation. Efforts to prevent or slow the spread of nuclear weapons and the materials and technologies used to produce them.

Normal operation. All normal conditions and those abnormal conditions that frequency estimation techniques indicate occur with a frequency greater than 0.1 events per year.

Nuclear fuel. Materials that are fissionable and can be used in nuclear reactors.

Plutonium. A manmade fissile element. Pure plutonium is a silvery metal that is heavier than lead. Material rich in the plutonium-239 isotope is preferred for manufacturing nuclear weapons, although any plutonium can be used. Plutonium-239 has a half-life of 24,000 years.

Population dose. See collective dose.

Probable maximum flood. The largest flood for which there is any reasonable expectancy in a specific area. The probable maximum flood is normally several times larger than the largest flood of record.

Processing (of spent nuclear fuel). Applying a chemical or physical process designed to alter the characteristics of the spent nuclear fuel matrix.

Public. Anyone outside the DOE site boundary at the time of an accident or during normal operation.

PUREX. An acronym for Plutonium-Uranium Extraction, the name of the chemical process usually used to reprocess spent nuclear fuel and irradiated targets.

Rad. The special unit of absorbed dose. One rad (0.01 gray) is equal to an absorbed dose of 100 ergs/gram.

Radiation (ionizing). Energy transferred through space or other media in the form of particles or waves. In this document, we refer to ionizing radiation which is capable of breaking up atoms or molecules. The splitting, or decay, of unstable atoms emits ionizing radiation.

Radioactive waste. Waste that is managed for its radioactive content; solid, liquid or gaseous material that contains radionuclides regulated under the AEA of 1954, as amended and of negligible economic value considering costs of recovery.

Radioactivity. The spontaneous emission of radiation from the nucleus of an atom. Radionuclides lose particles and energy through this process of radioactive decay.

Region of influence. Region in which the principal direct and indirect socioeconomic effects of actions are likely to occur and are expected to be of consequence for local jurisdictions.

Regulated substances. A general term used to refer to materials other than radionuclides that may be regulated by other applicable Federal, State, (or possibly local) requirements.

rem. Roentgen Equivalent Man which is a unit of dose equivalent. Dose equivalent in rem is numerically equal to the absorbed dose in rad multiplied by a quality factor, distribution factor and any other necessary modifying factor (1 rem = 0.01 sievert).

Reprocessing (spent nuclear fuel). See chemical separation.

Risk. Quantitative expression of possible loss that considers both the probability that a hazard causes harm and the consequences of that event.

Saltstone. Low-radioactivity fraction of high-level waste formed into a concrete block at the Savannah River Site.

Source material. (1) Uranium, thorium, or any other material that is determined by the Nuclear Regulatory Commission pursuant to the provisions of the Atomic Energy Act of 1954, Section 61, to be source material; or (2) ores containing one or more of the foregoing materials, in such concentration as the Nuclear Regulatory Commission may by regulation determine from time-to-time [Atomic Energy Act 11(z)].

Special nuclear material. (1) Plutonium, uranium enriched in the isotope 233 or in the isotope 235, and any other material that the Nuclear Regulatory Commission, pursuant to the provisions of the Atomic Energy Act of 1954, Section 51, determines to be special nuclear material, but does not include source material; or (2) any material artificially enriched by any of the foregoing, but does not include source material.

Spent nuclear fuel. Fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated.

Stabilization (of spent nuclear fuel). Actions taken to further confine or reduce the hazards associated with spent nuclear fuel, as necessary for safe management and environmentally responsible storage for extended periods of time. Activities which may be necessary to stabilize spent nuclear fuel include canning, processing, and passivation.

Storage. The collection and containment of waste or spent nuclear fuel in such a manner as not to constitute disposal of the waste or spent nuclear fuel for the purposes of awaiting treatment or disposal capacity (i.e., not short-term accumulation).

Surface water. All waters that are open to the atmosphere and subject to surface runoff. All waters naturally open to the atmosphere (rivers, lakes, reservoirs, streams, impoundments, seas, estuaries, etc.) and all springs, wells, or other collectors that are directly influenced by surface water.

Target. A tube, rod, or other form containing material that, on being irradiated in a nuclear reactor would produce a designed end product (i.e., uranium-238 produces plutonium-239 and neptunium-237 produces plutonium-238).

Target material. Residual material that is left after a target has been irradiated and dissolved, and the end product has been removed. In this EIS, target material contains enriched uranium and fission products.

Total effective dose equivalent. The sum of the deep dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).

Type B packaging. Packaging for radioactive material which meets the standards for Type A packaging and, in addition, meets the standards for the hypothetical accident conditions of transport as prescribed in 49 Code of Federal Regulations Part 173.398(c). This includes spent fuel casks.

Ultimate disposition. The final step in which a material is either processed for some use or disposed of.

Undiscounted dollars. Expressing income and expenditures in the year they occur, not at some common point in time.

Uranium. The basic material for nuclear technology. It is a slightly radioactive naturally occurring heavy metal that is more dense than lead. Uranium is 40 times more common than silver.

Vitrification. The process of immobilizing waste that produces a glass-like solid that permanently captures the radioactive materials.

Vulnerabilities. Conditions or weaknesses that may lead to radiation exposure to the public, unnecessary or increased exposure to the workers, or release of radioactive materials to the environment.

Waste classification. Wastes are classified according to 10 CFR § 61.55 for the purpose of near surface disposal to three classes: A, B, and C. Class C waste represents the waste that must meet the most rigorous requirements on waste form to ensure stability and additional measures at the disposal facility to protect against inadvertent intrusion.

Waste management. The planning, coordination, and direction of those functions related to generation, handling, treatment, storage, transportation, and disposal of waste, as well as associated surveillance and maintenance activities.

Waste minimization. An action that economically avoids or reduces the generation of waste by source reduction or reduces the toxicity of hazardous waste, improving energy usage, or by recycling. This action will be consistent with the general goal of minimizing present and future threats to human health, safety, and the environment.

Wet storage. Storage of spent nuclear fuel in a pool of water, generally for the purposes of both cooling and worker shielding.